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Report of the U.S. Nuclear Regulatory Commission Piping Review Committee

Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants

U.S. Nuclear Regulatory Commission

Prepared by The Pipe Crack Task Group

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants

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Prepared by
The Pipe Crack Task Group

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TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

FOREWORD

The Executive Director for Operations of the U. S. Nuclear Regulatory Commission (NRC) requested that a comprehensive review be made of NRC requirements in the area of nuclear power plant piping. In response to this request an NRC Piping Review Committee was formed. The activities of this Review Committee were divided into four tasks handled by appropriate Task Groups. These were:

- Pipe Crack Task Group
- Seismic Design Task Group
- Pipe Break Task Group
- Dynamic Load/Load Combination Task Group

Each Task Group will prepare a report appropriate to its scope. In addition the Piping Review Committee will prepare an overview document rationalizing areas of overlap between the Task Groups. This will be released as a separate report.

Because of the nature of the current intergranular stress-corrosion cracking (IGSCC) problems in boiling water reactors (BWRs), the Pipe Crack Task Group is on an accelerated schedule. The first draft of this report was completed March 1984, while the other Task Groups are aiming for August - September. The Review Committee is scheduled to complete its activities prior to the end of 1984.

The project titles of the five volumes are:

Volume I - Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants

Volume II - Evaluation of Seismic Designs

Volume III - Evaluation of Potential for Pipe Breaks

Volume IV - Evaluation of Other Dynamic Loads and Load Combinations

Volume V - Summary - Piping Review Committee Conclusions and Recommendations

ACKNOWLEDGMENT

The Task Group on Pipe Cracking wishes to thank those who assisted in the preparation of this report. The Task Group members, consultants, and major contributors are listed in Appendices A and B. Others who assisted in preparation of individual sections included: from the utility industry, J. Sinclair (Vermont-Yankee) R. Godby (Hatch), W. Pierce and M. Tucker (Dresden), and J. McElroy (PECO); from NRC, D. Clark, G. Gear, M. Boyle, and R. Stark; from ANL, L. Beckford; from PNL, G. Selby; from BNL, C. Auerbach and C. Czajkowski; also J. Spanner of Spanner Engineering.

Our thanks to Brookhaven National Laboratory for its support in the use of its facilities and staff to prepare the report, particularly to Ms. Olga Betancourt for typing coordination and Ms. Mary Rustad for editorial assistance and report coordination.

The Task Group met with representatives of EPRI, General Electric Co., BWR Owners Group, NRC-RES, utilities, ASME code groups, and personnel in foreign countries.

A special international team of experts reviewed and commented on a draft version of this report, consisting of the following:

Dr. Yoshio Ando	(Japan)
Dr. Ferenc de Kazinczy	(Sweden)
Dr. Karl Kussmaul	(Federal Republic of Germany)
Dr. Brian Tomkins	(United Kingdom)

Their comments appear as Appendix H.

EXECUTIVE SUMMARY

The Executive Director for Operations concurred in the recommendations of the Committee for Review of Generic Requirements (CRGR) that the "NRC should develop an integrated program plan to deal with the entirety of the stress corrosion cracking problem in BWR piping in order to avoid a piecemeal approach to the problem. Such a program should enlist the aid of recognized experts in the area and result in a coordinated resolution that takes into consideration the entire spectrum of BWR plants."

The Task Group on Pipe Cracking has attempted to comply with this charter. The title of the report mirrors the scope of the study, namely, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants." A review of cracking incidents in both BWRs and PWRs since the release of NUREG-0531 (primarily BWRs) in 1979 and NUREG-0691 (PWRs) in 1980 reveals that incidents due to all types of cracking for both BWRs and PWRs illustrate no unusual trends other than IGSCC in the recirculating system of BWRs (Section 3). Therefore, as noted in Section 1, the Task Group on Pipe Cracking made a conscious decision to limit its scope to IGSCC in BWRs.

In essence the overall scope of this document is quite comparable to that of NUREG-0531 with one notable exception. The Task Group has made a conscious effort to relate recommendations to regulatory documents that would require changes if such recommendations are implemented. In fact, a comparison of the relevant conclusions and recommendations of NUREG-0531, which are repeated in Appendix I of this report, with those of this report confirms the strong similarity of the conclusions and recommendations between the two documents.

Stress-corrosion cracking, particularly intergranular stress-corrosion cracking (IGSCC), is not a new phenomenon in nuclear reactor piping. For example, we have experienced IGSCC in a range of piping sizes in BWRs over the past 25 years. To our knowledge the most severe leaking crack was at or below the technical specification limits; therefore high leak rates have not been a problem and there have been no structural failures induced by IGSCC. The fact that the current problem affects the larger recirculation piping has resulted in increased concern.

As indicated, the Task Group on Pipe Cracking considers that IGSCC is a serious problem requiring some changes to current regulatory practice. However, we do not feel that IGSCC represents such an urgent problem that it necessitates immediate additional regulatory action. The ongoing BWR surveillance and monitoring programs are adequate short-term responses to the problem.

Both value-impact and risk studies indicate that pipe failures, even assuming the higher rates due to IGSCC utilized in this report, are not a major contributor to core melt. However, we believe that the following recommendations represent good engineering practice that should markedly reduce levels of inservice inspection and reactor outage time.

As discussed in Section 2, IGSCC occurs because of a synergism between the three causative factors: sensitization, an environment conducive to initiation and propagation of IGSCC, and high tensile stresses (primarily residual, but other loads also are contributors). The Task Group recommends that at least two of the three causative factors need to be modified to minimize IGSCC, although best practice calls for modification of all three. The preferred action to combat IGSCC is to replace materials such as Types 304 and 316 with a resistant material such as Type 316NG stainless steel. Recommended measures are summarized in the table at the end of this Executive Summary.

With regard to replacement material, control of carbon to appropriately low levels, e.g., less than 0.03, appears acceptable and has been used in the past; the Task Group recognizes that this results in an ASME Code penalty because of the lower strength of the low carbon stainless steels, which requires thicker pipe for the same circumstances. A preferable alternative is an alloy such as Type 316NG where the carbon is low enough that the material is highly resistant to IGSCC and the strength is retained through controlled additions of nitrogen. While Type 316NG is more resistant to IGSCC than Types 304 and 316, greater care must be exercised in using it to limit hot short cracking by placing controls on composition and fabrication variables related to hot forming and welding.

The control of water chemistry is discussed in Section 6. The Task Group believes that conductivity control to a low level is possible and beneficial, but that elimination of IGSCC in piping systems containing a sensitized stainless steel will require the use of hydrogen additions to lower the electrochemical potential of the steel to a nonsusceptible level. The Task Group recognizes the limited nature of the work in this area. Even so, it reiterates its recommendation cited here and in NUREG-0531 that water chemistry should be improved.

The Task Group recognizes there are several measures capable of reducing the tensile residual stress level in the heat-affected zone (HAZ). With appropriate control of parameters, a number of these processes should be acceptable; however, the group has a definite preference for induction heating stress improvement (IHSI), based on a review of the extent and applicability of the available data.

A major problem cited in NUREG-0531 was the inability of ultrasonics to detect and size IGSCC in austenitic alloys reliably. Substantial progress has been made in the reliability of detection, assuming proper operator training and procedures; however, more work is required to develop a better understanding of the variability from operator to operator. The existing techniques are adequate to size the crack length to acceptable limits of accuracy. With regard to crack depth sizing, the techniques as currently practiced are clearly inadequate and require more work. Sections 4 and 5 discuss the problem at length as well as specific measures taken or to be taken to improve the reliability of crack detection and sizing.

A substantial percentage of welds are limited to single-sided access. This will remain the case even after replacement of recirculation systems with a material such as 316NG if the original pumps, valves, vessel nozzles, etc., are used and their regions adjacent to the weld are too short for UT. Attention to stress reduction with IHSI should be considered for these components. Every effort should be made to improve accessibility and weld joint design for ultrasonic testing (UT) when pipes are replaced.

The changes in overall LOCA risk because of IGSCC and a value-impact analysis of the options available to the utility are discussed. Both studies arrived at the same major conclusion, namely, that pipe breaks, even when higher rates are assumed as a result of IGSCC, represent a minor contribution to the frequency of core melt. A deterministic study discussed in Section 6 suggests a similar conclusion; leak-before-break is the dominant mode rather than catastrophic failure.

The preceding has been a broad overview of the current problem of IGSCC in BWR recirculating lines and the recommendations of the Task Group on Pipe Cracking. Chapter 9 pulls together all the conclusions and recommendations throughout the document. The following are conclusions and recommendations considered to be most significant by the Task Group:

- Because mitigating actions addressing only one of the factors may not be fully effective under all anticipated operating conditions, mitigating actions should address two and preferably all three of the causative factors; e.g., material plus some control of water chemistry, or stress reversal plus controlled water chemistry.

- Code minimum UT procedures result in totally inadequate IGSCC detection. Easily implementable modifications to these procedures have resulted in some improvement. These have been incorporated into Code Case N-335. Therefore, it is recommended that Code Case N-335 should immediately be made mandatory for all augmented inspections until better procedures are developed.

- Although IGSCC detection has improved to the point that it is considered acceptable under optimum conditions and procedures, the detection reliability as impacted by variability in operator procedure and equipment performance along with field conditions needs further study and improvement. While length sizing of cracks is acceptable, depth sizing is currently inadequate. It is recommended that advanced techniques and procedures for crack detection and depth sizing continue to be developed and incorporated into Code requirements to provide data to reduce the need for extremely conservative fracture mechanics evaluation.

- The current activities in personnel and procedure qualification and performance demonstration represent steps in the right direction, and the resultant process that is being implemented is acceptable in the interim; however, they need further improvement. Therefore, it is recommended that ongoing industry and NRC activities to develop adequate criteria for qualification of the entire inspection process to achieve more reliable field inspection be completed and implemented on a high priority basis.

- For future plants or for replacement of existing piping systems, the material, design of pipe joints, and accessibility from both sides of the weld should be optimized for UT examinations; this requirement should be mandatory for all components with the exception of existing items such as pumps, valves and vessels in older plants. The uninspectable joints should be subjected to IHSI.

- Inspection techniques should be developed for detection and dimensioning of flaws in pipes repaired by the weld overlay process.

- All Type 304 and 316 austenitic piping systems operating over 200°F (93°C) should receive augmented inservice inspection unless they have been treated with effective countermeasures.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair. The limitation on acceptable crack length is primarily a result of the lack of confidence in flaw depth sizing capability, and is intended to ensure leak-before-break conditions. The maximum allowable throughwall crack length can be determined based on weld joint specific loads.

- The maximum crack length allowable without repair for a specific weld joint should be the minimum of either 1) the throughwall crack length demonstrated by elastic-plastic fracture mechanics analyses to be stable under normal operating plus SSE loading conditions, 2) the throughwall crack length that would still permit the pipe to withstand normal operating plus SSE loading conditions as demonstrated by net section collapse (limit-load) analyses, or 3) the maximum crack length that would result in a leak rate greater than the plant's normal makeup capacity. Shorter cracks can be evaluated using the IWB-3640 criteria as modified by the NRC staff in SECY 83-267C. Calculations indicate that in the majority of cases the maximum crack length acceptable under the above criteria will be approximately 25% to 30% of the pipe circumference.

- On the basis of fracture mechanics evaluation for bounding and typical stress conditions and weld toughness properties, it is concluded that IWB-3640 provides an adequate basis for evaluating the majority of the weld connections in BWR recirculation piping. This is especially true because many of the cracks will be in higher toughness zones adjacent to the lower toughness welds.

- Additional fracture mechanics analyses, material properties characterization, and large scale pipe tests be performed to understand further the implications of stainless steel weld and cast material fracture toughness properties in flawed pipe evaluations. Furthermore, in this regard, the Task Group recommends active NRC support of the ASME Task Group currently evaluating the concerns which have been raised regarding IWB-3640.

- Since operating experience and fracture mechanics evaluations indicate that leak-before-break is the most likely mode of piping failure, it is recommended that reasonably achievable leak detection procedures be in effect in operating plants. Current sump pump monitoring systems are sensitive enough to provide additional margin against leak-before-break if more stringent requirements on surveillance intervals and unidentified leakage are imposed (see Section 4). Therefore, the Task Group recommends that the limits on unidentified leakage in BWRs be decreased to 3 gpm and that the surveillance interval be decreased to 4 hours or less.

- For relatively short axial cracks, analysis can be used to justify long-term operation with weld overlays, since errors on crack depth measurement or flaw growth predictions for these cracks will lead at worst to relatively small leaks, which will be easily detectable long before the crack can grow long enough to cause failure. For circumferential cracks weld overlay is considered an acceptable repair procedure for a maximum of two refueling outages unless reliable techniques for the sizing of cracks through the overlay or for the monitoring of crack growth are developed.

- Experience with materials to mitigate IGSCC, such as 347NG in Germany and 304NG and 316NG in Japan has been excellent. Other materials used in the U. S. include 304L and 316L. All of these alloys are more resistant to IGSCC than conventional Types 304 or 316 stainless steel. Based on U. S. data and prior use, Type 316NG stainless steel offers an additional margin of resistance to IGSCC and utilities that choose to replace pipe should be strongly encouraged to use it.

- Although low-carbon stainless steels with nitrogen additions have been successfully fabricated and welded in Japan and Europe, U. S. experience with these materials is limited. It appears that greater care must be exercised in the control of composition and fabrication variables to limit cracking during hot forming or welding.

- The use of IHSI on weldments with detectable cracking must be considered on a case-by-case basis. However, for relatively short cracks (approximately 20% of the circumference in length), since even large errors in crack sizing or the prediction of flaw growth will lead only to small leakage, the decision on whether an additional repair is required can be determined by analysis. For longer cracks, repair will probably be required.

- IHSI is considered to be a more effective mitigating action for IGSCC than HSW and LPHSW in part because more data are available to demonstrate that the process does produce a more favorable residual stress state. All the residual stress improvement remedies are considered to be much more effective when applied to weldments with no reported cracking.

- BWR water chemistry controls should be modified to minimize IGSCC. These modifications should include both a substantial reduction in the levels of ionic species entering the primary coolant and a control of oxygen level. The current work on reduction of oxygen through hydrogen additions should be followed closely with the possibility that it may be employed to reduce further the electrochemical potential of the stainless steel to a level at which SCC, either IGSCC or TGSCC, will not occur. It appears that hydrogen water chemistry is an effective IGSCC countermeasure. However, ongoing work regarding potential adverse effects on other reactor components should be closely followed in order to confirm the acceptability of this countermeasure.

- The Task Group recommends that the inspection schedule for welds be based on the resistance to IGSCC of the materials and the effectiveness of the mitigating processes applied to the welds. The materials which are considered resistant and the categories of countermeasures processes are:

Resistant Materials

(1) 304L, 316L, 316K, 304NG, 316NG, 347NG, 308L
(2) low-strength carbon steels
(3) approved nickel-based materials
(4) cast low-carbon/high-ferrite austenitic stainless steels
(5) welds solution heat-treated after fabrication and welding.
(6) other, as approved by NRC

Countermeasure A Processes - Any combinations of two mitigating processes which are intended to reduce or minimize any two of the three causes (i.e., sensitization, stress and environment) contributing to IGSCC. These are summarized below:

(1) IHSI on new pipe or pipe with no reported indications plus hydrogen water chemistry*;

(2) HSW on new welds, including new welds used to install short repair sections ("pup pieces") plus hydrogen water chemistry;

(3) LPHSW on new pipe welds including new welds used to install short repair sections plus hydrogen water chemistry.

(4) Other, as approved by NRC.

<u>Countermeasure B Processes</u> - Any of the following mitigating processes.

(1) IHSI on new pipe or pipe with no reported indications.

(2) HSW on new welds including new welds used to install short repair sections ("pup pieces").

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*Hydrogen water chemistry is discussed in Section 6.

(3) LPHSW on new pipe welds including new welds used to install short repair sections.

(4) Hydrogen water chemistry.

(5) Other, as approved by NRC.

The recommended inspection requirements are:

Condition and Inspection Required for Weld Categories

Weld Category	Condition	Inspection Required
A	Resistant material or countermeasure A applied	25% of the welds of each pipe size in 10 years. At least one-third of these should be inspected every three and one-third years or the nearest refueling outage.
B	Nonresistant material with countermeasure B applied	50% of the welds of each pipe size in 10 years. At least one-third of these should be inspected every three and one-third years or the nearest refueling outage.
C	Neither of the above, and all welds with detectable cracking regardless of the use of mitigating processes	100% in 6 years. At least one-half of these should be inspected every three and one-third years or the nearest refueling outage.
		For plants older than 6 years, all uninspected Category C welds shall be inspected at the next outage after compliance with IE Bulletin 82-03 or 83-02.*

*Weld inspections not performed in accordance with IE Bulletin 82-03 or 83-02 are considered inadequate; therefore, no credit is given for such inspections.

Recommended Measures for Controlling IGSCC in BWR Piping

	Near Term			Long Term		Ultrasonic Exam.			
BWR Plant Status	Overlay Weld*	Replace with Similar Alloy	Residual Stress Improvement IHSI, HSW	Hydrogen Water chemistry	316 NG Piping,	New Baseline	Accel. UT	Norm. UT	Enhanced Leak Detection (moisture tapes, AE, etc.)
Design Stage			X	X	X			X	
NTOL or Recent Startup			X	X		X		X	
Operating >5(?) years									
No Cracks Detected			X	X		X	X------	X#	X (maybe)
Only Limited Shallow Cracks			X	X		X	X^{+}------	X	X
Deeper Cracks -- Few or Many	X	X maybe	X**	X	X	X	X then	X (for 316NG)	X Prior to Replacement

*Limited to two cycles unless convincing evidence is presented

**Also suggested after replacement with 316 NG

Prior to mitigation, accelerated UT; thereafter normal UT

X Accelerated UT limited to cracked welds

+ With mitigation, accelerated UT limited to cracked welds

1.0 INTRODUCTION

1.1 PURPOSE

On August 1, 1983, the Executive Director for Operations of the Nuclear Regulatory Commission (NRC) established a committee to review nuclear piping in the context of current regulations, regulatory guides, branch technical positions, and other pertinent documents. A Task Group of this committee was assigned the responsibility for reviewing specific incidents of cracking in both boiling water reactors (BWRs) and pressurized water reactors (PWRs) with emphasis on intergranular stress-corrosion cracking (IGSCC) reported in 1982-1983 in the BWR recirculating systems. The charter of the Task Group included a spectrum of issues such as:

- The significance of IGSCC in the context of overall reactor safety;
- the reliability of detection and of sizing of IGSCC in BWR piping using current ultrasonic techniques;
- the potential effects of changes in water chemistry on the initiation and propagation of IGSCC;
- the value and/or limitations of repair techniques such as overlay cladding on piping containing IGSCC;
- the long-term value of alternative measures such as induction heating stress improvement (IHSI) on both uncracked and cracked piping;

- the long-term reliability of replacement piping such as 316NG and the possible limitations of the material;

- a review of information pertinent to pipe cracking occurring in foreign light water reactors.

Previous Pipe Crack Study Groups have reviewed IGSCC in BWRs [Technical Report--Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants, NUREG-75/067 (1.1) and in LWRs Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531 (1.2)]. A Pipe Crack Study Group also reviewed incidents of cracking in PWRs (1.3).

One Pipe Crack Study Group (1.2) reviewed and commented on the significance of IGSCC in large-diameter piping typical of BWR recirculating lines. Their conclusion is given below:

- The cracking experience in a German BWR facility demonstrates that IGSCC can occur in large-diameter (more than 20 in.) BWR stainless steel piping and confirms the previous study group's conclusions that, although cracking in large-diameter pipes has not been observed in the United States, the conditions for IGSCC do exist in these lines. We also have reviewed the safety aspects associated with the potential for IGSCC in large-diameter BWR stainless steel piping. Based on our review, we have concluded that, although throughwall and part-through wall IGSCC can exist in BWR stainless steel piping during plant operation, it is unlikely that significant cracks would go undetected. In addition, based on our review and analysis, we also conclude that it is unlikely these cracks will cause unstable crack growth and excessive loss of coolant. Further, since the Emergency Core Cooling System (ECCS) provides protection should a loss-of-coolant accident occur, we conclude that IGSCC in these lines, while generally undesirable, will not be a hazard to public health and safety.

Several of the conclusions and recommendations of reference 1.2 are particularly pertinent to this study. Therefore, the relevant conclusions and recommendations from NUREG-0531, numbered as in the report, are repeated as Appendix I to this report, to serve as "benchmarks" throughout this document as to progress or lack of progress since 1979.

1.2 REVIEW OF PRESSURIZED WATER REACTORS

Pressurized water reactor (PWR) piping was reviewed cursorily in 1979 (1.2) and it was concluded that stress corrosion was not a problem for the primary system. A more extensive study in 1980 (1.3) examined several failure mechanisms relevant to PWR piping systems. Again stress corrosion was considered to be a secondary contributor where attack was limited to lower pressure, thin-wall secondary or tertiary systems.

PWR primary piping systems whether ferritic or austenitic have been essentially immune to any cracking mechanisms. In fact, probabilistic studies at Lawrence Livermore National Laboratory (LLNL) yield very small (10^{-12} to 10^{-14}) direct failure levels and acceptably small (approximately 10^{-8}) per reactor year failure levels by indirect failures such as cranes falling on piping during several seismic events. The LLNL studies and extensive studies in the Federal Republic of Germany (1.4) both led to the conclusions that catastrophic failure of PWR primary piping is an incredible event.

A paper by Bush (1.4) at an IAEA Symposium on <u>Reliability of Reactor Pressure Components</u> reexamined PWR piping failures through 1982. In addition, an unpublished report by Bush examined PWR failures through 1982 and partly into 1983. Both studies confirmed that there have been very few cases of SCC in any PWR piping systems during the period since reference 1.3.

Since 1980, failures due to vibrational fatigue, thermal fatigue, and corrosion fatigue have been quite limited. Corrosion-cavitation has resulted in severe failures; however, these have not been in safety systems, e.g., the large steam turbine lines. The one possible exception has been in the area of dynamic loads such as water hammer. There has been a feedwater line failure at Maine Yankee which is understandable because steam generator J-tubes had not been installed to prevent such incidents.

On the basis of the preceding comment, the Task Group has concluded that there are no bases for an extensive review and analysis of PWR pipe cracking. With regard to inspectability, Section 4 discusses the relative reliability of NDE of PWR primary materials.

1.3 DOCUMENTS POTENTIALLY AFFECTED BY RECOMMENDATIONS IN THIS REPORT

The Task Group was directed in its instructions to cite various documents that might require changes as a result of the recommendations developed in this report. The following is a citation of these documents. Specific suggestions concerning changes appear in the appropriate sections:

Generic Issues

A-14	Flaw Detection
A-42	Pipe Cracks in BWRs (PGI-8)
No. 34	Reactor Coolant Systems Leakage

Regulations

10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix A General Design Criteria for Nuclear Power Plants

Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Criterion 36 - Inspection of Emergency Core Cooling System

Regulatory Guides

1.31 - Control of Ferrite Content in Stainless Steel Weld Metal

1.44 - Control of the Use of Sensitized Stainless Steel

1.45 - Reactor Coolant Pressure Boundary Leakage Detection Systems

1.46 - Protection Against Pipe Whip Inside Containment

1.56 - Maintenance of Water Purity in Boiling Water Reactors

1.58 - Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

1.84 - Code Case Acceptability in ASME Section III Design and Fabrication

1.85 - Code Case Acceptability in ASME Section III Materials

1.116- Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

1.130- Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports

1.147- Inservice Inspection Code Case Acceptability ASME Section XI Division 1

NUREG-0313 - Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, Rev. I, 1979, Rev. II, in draft form

NUREG-0800 - Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants

3.6.1 - Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

3.6.2 - Determination of Rupture Locations and Dynamic Effects Associated

5.2.1.1 - Compliance with the Codes and Standards Rule 10 CFR 50.55a

5.2.1.2 - Applicable Code Cases

5.2.3 - Reactor Coolant Pressure Boundary Materials

5.2.4 - Reactor Coolant Pressure Boundary Inspection and Testing

5.2.5 - Reactor Coolant Pressure Boundary Leakage Detection

Codes and Standards

ASME XI - UT

ANSI Draft Standard - Leak Detection

1.4 THE TASK GROUP ON IGSCC

The Task Group is composed of ten members from the NRC and one member from Review and Synthesis Associates, listed in Appendix A. The ten members represent the Offices of Nuclear Reactor Regulation (4), Nuclear Regulatory Research (3), Inspection and Enforcement (1), and one each in Regions 1 and 2. Services of several additional NRC staff members (cited in the Acknowledgements) were used as required, so that the expertise of the personnel in the division was factored into the investigation of the Task Group.

In addition, the services of outside consultants in the fields of corrosion chemistry, nondestructive examination, and fracture mechanics were used. They are listed in Appendix B.

The Task Group limited its scope to IGSCC of BWR piping on the basis of its charter and the arguments raised in Section 1.2 regarding PWRs.

The Task Group met with representatives of EPRI, General Electric Company, BWR Owners Group, NRC-RES, utilities, ASME code groups, and personnel in foreign countries.

1.5 REFERENCES

1.1 Pipe Crack Study Group. 1975. Investigation and Evaluation of Cracking in Austenitic Stainless-Steel Piping of Boiling Water Reactor Plants. NUREG-75/067, U.S. Nuclear Regulatory Commission, Washington, D.C.

1.2 Pipe Crack Study Group. 1979. Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants. NUREG-0531, U.S. Nuclear Regulatory Commission, Washington, D.C.

1.3 PWR Pipe Crack Study Group. 1980. Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors. NUREG-0691, U.S. Nuclear Regulatory Commission, Washington, D.C.

1.4 S. H. Bush. 1983. "Pressurized Water Reactors." In Proceedings of Reliability of Reactor Pressure Components, pp. 29-56, IAEA-SM-269/73, International Atomic Energy Agency.

2.0 CAUSES AND DESCRIPTION OF IGSCC PHENOMENA

As noted in the introduction, intergranular stress-corrosion cracking (IGSCC) has been occurring in BWR and PWR nuclear power plants for a number of years. The recent identification of the problem in a number of operating BWRs has not indicated the occurrence of any new phenomenon, but rather that the phenomenon already known and discussed in the previous review group reports is being found in a wider range of locations, probably because of longer operating times and improved inspection and detection techniques. Further, as a result of the considerable research performed in this area since our previous reviews, our understanding of the causes of IGSCC has improved. In this section, therefore, the purpose is to summarize our current understanding of the problem in order to provide a background for the discussions in later sections on short- and long-term solutions and remedial actions.

IGSCC in stainless steels arises from the synergistic effects of stress and the environment on a material that is susceptible to the phenomenon. Each of these aspects of the synergism will be examined separately.

A susceptible material: Types 304 or 316 SS, in a certain temperature range, will form grain boundary networks of Cr carbides, accompanied by adjacent areas depleted in Cr. The production of the Cr carbides in these areas results in a network of narrow zones adjacent to the grain boundaries in which the Cr levels in solid solution are sufficiently lower than those in the bulk of the grains to enable electrochemical cells to be set up between the lower Cr grain boundary material and the higher Cr bulk grain material. Precipitation of the Cr carbides and formation of the Cr-depleted zones are a function of the composition of the steel and the time that the steel is heated in the temperature range at which the kinetics of the reaction are sufficiently rapid and thermodynamics favor production of the Cr carbides. Typically, this occurs between 900 and 1500°F, although some low temperature sensitization (LTS) can be anticipated at temperatures as low as the operating temperatures of a BWR. While sensitization can be produced in stainless steels containing carbon levels as low as 0.02 wt.%, IGSCC has seldom if ever been seen in BWRs in stainless steels containing significantly less than 0.04 wt.% carbon; the probability of IGSCC occurring appears to increase rapidly with carbon concentrations above this level up to 0.055%, and thereafter to be constant up to the limit for Type 304 or Type 316 SS of 0.08 wt.%. All stainless steels are specified by the industry and by NRC positions to be in the solution-annealed state prior to fabrication. At the solution-annealing temperatures, i.e., at 2000°F or higher, the Cr carbides are unstable with respect to solid solutions of Cr and carbon in the stainless steel, and thus decompose. Rapid cooling following solution-annealing prevents recombination from occurring. However, the welding of stainless steel pipe produces a narrow band of material in an area adjacent to the weld which has been heated into the sensitization range by the heat of welding; the pipe cracks are occurring primarily in this area.

Although the general phenomenon of IGSCC has been mainly attributed to the formation and extent of the Cr-depleted zones in the grain boundaries, other minor constituents of the stainless steel can affect either the kinetics of formation of the Cr-depleted zones or the initiation and propagation of the stress-corrosion cracks, as discussed below. Steels containing alloying elements that form carbides more stable than Cr carbide (such as niobium carbide or Ti carbide) are less prone to sensitization. However, even with these steels there is a potential area immediately adjacent to the weld fusion line in which Cr-depleted zones can be produced if the carbon levels are high (knife-line attack). The relative freedom of the Swedish and the newer German BWRs from IGSCC is thought to be related to the use of a 0.05% carbon limit in Sweden, and the use of a low-carbon, niobium-stabilized stainless steel (often referred to as Type 347NG) in West Germany.

Stress: Austenitic stainless steels as a general rule do not deform elastically to a sharp yield stress and plastically at stresses above it. Rather, some plastic deformation can occur at stresses considerably below the nominal "design yield" (0.2% offset) strength. It is generally considered that the straining or creep of the stainless steel under tension produces a rupture in the protective oxide films, followed by a relatively rapid bare metal corrosion until the film reheals. The lower Cr content in the sensitized grain boundaries results in locally greater metal loss prior to rehealing. Repeated rupture of the film, caused either by alternating the stresses or applying continuous stresses appreciably above the yield point, produces a crack along the grain boundary, since preferential attack occurs at the Cr-depleted grain boundaries. The primary source of tensile stress in the operating BWR piping is considered to be the residual stresses from welding, as discussed by the earlier review groups. Added to this, however, are operational pressure stresses, vibrational stresses, and thermal stresses, particularly during heatup and cooldown. The combined effects of all these tensile stresses on the inner surface of the piping in the sensitized regions need to be considered in determining the susceptibility of a given welded area to IGSCC. Generally, sufficient stress to cause repeated rupture of the protective oxide film is necessary to initiate cracking. Although in terms of conventional engineering design, creep is negligible in austenitic stainless steels at BWR operating temperatures, the strain rates associated with stresses at or above "yield" are sufficient to produce rupture of the oxide film. This may even be true for stresses below "design yield," particularly if they are applied intermittently, because of the tendency of the material to creep. Once a crack is initiated, a continued tensile stress intensity at the crack tip is required for crack propagation.

Environment: In the BWR primary coolant, the presence of residual oxygen from radiolysis puts the chemical potential of the stainless steel in a range in which the intergranular stress corrosion cracks can initiate and propagate in sensitized material. Oxygen appears to establish a chemical potential such that the electrochemical crack propagation process can proceed and is therefore the key environmental cause of IGSCC. The species responsible for conducting the electrochemical currents in IGSCC are assumed to come either from the environment (such as in-leakage of impurities from the condenser or resin bead breakup and decomposition) or from corrosion of the metal itself, particularly from bare-metal corrosion following film rupture, producing salts of sulfur, phosphorus, silicon, or other anions. These species can produce a current-carrying capability in the grain boundary cracks, so that IGSCC can occur in theoretically pure water at sufficiently high stresses in the presence of an oxidizing potential.

2.1 MECHANISMS OF IGSCC

IGSCC can be described in terms of an initiation phase and a propagation phase, the mechanisms of which are reviewed briefly below.

2.1.1 Initiation Phase

The surface of stainless steels is protected from corrosion by a thin, adherent, passivating protective oxide. In addition to this passive layer, the surfaces of the austenitic stainless steel pipes in a BWR have a relatively thick layer of mass-transported oxides deposited on them in the form of crud, which contains some radioactivity. These crud deposits can also be highly specific adsorbents for impurities entering the solution, such as sulfates or chlorides, and this property may have some influence on the crack-initiation process. It is difficult to extrapolate laboratory data to determine the precise point of crack initiation. In slow strain rate stress-corrosion tests, if the protective passivating oxide layers are ruptured frequently enough, localized corrosion of sensitized grain boundary material at the root of these ruptures can produce an environment in a relatively short period of time that is capable of carrying the electrochemical corrosion current, and therefore bringing about IGSCC. The presence of the crud deposits and of materials or ions adsorbed in these deposits may also influence the initiation of IGSCC. Laboratory tests have shown that very small amounts of sulfates and chlorides can accelerate the initiation of IGSCC in a constant load test when the electrochemical potential is held in a range at which IGSCC is known to occur.

Crack initiation is assumed to occur at the point at which an electrolyte builds up in the crevice between the ruptured protective oxide and the bare-metal surface. This rather vaguely defined point of transition between initiation and propagation reactions is difficult to measure experimentally. In terms of a mechanism for IGSCC, however, it is consistent with the experimental data.

2.1.2 Propagation

Once IGSCC has been initiated, the subsequent growth of these cracks depends upon several factors: the stress intensity (determined from the geometry of the crack and the applied tensile stress), the presence in the crevices of impurities that can stabilize a conducting (usually acidic) environment in the crevice, and the degree of sensitization of the material (which affects the potential of the electrochemical cells existing between the higher Cr and lower Cr areas). It also depends upon the continued presence in the bulk solution of an oxidizing species, controlling the overall chemical potential of the steel surface.

Experimental evidence on the role of impurities, electrochemical potential, sensitization, and stress intensity factors in crack propagation rates is much more abundant than are the data on crack initiation. This evidence has shown that crack propagation can be terminated by a shift in the electrochemical potential (i.e., by removing oxygen from the system), by a change in the material (i.e., by the crack running into a duplex weld structure containing sufficient delta ferrite to control the chromium-depleted zones), or by a reduction in the stress intensity factor (i.e., by the crack reaching an area of residual compressive stresses). Once a crack has been initiated, however, the environment within the crack is difficult to change, and will continue to favor propagation unless the chemical potential is changed by reducing the oxygen levels in the system. Thus, given sufficient stress intensity, a crack may continue to propagate into material normally considered resistant to IGSCC in a BWR environment.

2.2 CONCLUSIONS

- IGSCC in BWR piping is occurring owing to a combination of material, environment, and stress factors, each of which can affect both the initiation of a stress-corrosion crack and the rate of its subsequent propagation. In evaluating long-term solutions to the problem, one needs to consider the effects of each of the proposed remedial actions in terms of the current understanding of the initiation and propagation of stress-corrosion cracks.

2.3 RECOMMENDATIONS

- Mitigating actions to control IGSCC in BWR piping must be designed to alleviate one or more of the three synergistic factors: sensitized material, the conventional BWR environment, and high tensile stresses.

- Because mitigating actions addressing only one of the factors may not be fully effective under all anticipated operating conditions, mitigating actions should address two and preferably all three of the causative factors; e.g., material, plus some control of water chemistry, or stress reversal plus controlled water chemistry.

In the event that any of the preceding recommendations of this section are implemented, there may be a need to modify the following documents when relevant to the specific issue: NUREG-0313 - Rev. 2.

3.0 PRESENT STATUS OF PIPE CRACKING IN OPERATING BWRs

3.1 BACKGROUND

The Nine Mile Point Nuclear Station Unit 1 (a BWR Model 2) was shut down in March 1982 to replace recirculation pump seals. A normal hydrotest was performed to test the new seals. During this hydrotest, leaks were noticed at two of the furnace-sensitized Type 316 SS 28-in.-diameter recirculation loop safe ends. Further examination identified that the leaks originated in very small pinholes or cracks in the heat-affected zones of the safe-end-to-pipe welds. Subsequent ultrasonic examinations revealed extensive cracking at many weld joints in the Type 316 SS recirculation system. Two boat samples were removed for metallographic examination, and the cracking was identified to be IGSCC. The utility decided to replace the recirculation loops with IGSCC-resistant Type 316NG material.

This was the first known incident of IGSCC in large-diameter (larger than 10 inches) piping in the U. S., although cracks in large piping had been found in Japan, and cracks in 24-in. furnace-sensitized safe ends had been found in Germany.

After the extent of cracking at Nine Mile Point had been determined, IE Information Notice No. 82-39 was issued on September 21, 1982, to alert all BWR licensees to the problem. Meetings were held with General Electric, EPRI, and BWR Owners to discuss the relevance of the Nine Mile Point cracking to other BWRs.

On September 27, 1982, the NRC held a meeting with all BWR licensees to discuss plans for near-term inspections of welds in the large-diameter recirculation piping. This was followed on October 28, 1982, by IE Bulletin No. 82-03, "Stress Corrosion Cracking in Thick Wall, Large Diameter, Stainless Steel Recirculation System Piping at BWR Plants."

IE Bulletin No. 82-03 required that licensees of eight BWRs with scheduled outages through January 31, 1983, perform inspections of a reasonable sample of the welds in the recirculation system during the next outage. It further required that effective ultrasonic inspection procedures be used, and that the effectiveness of the procedures and competence of the UT examiners be demonstrated on samples of cracked piping from the Nine Mile Point recirculation system.

After these inspections showed that cracking in large pipes had occurred in five of the seven plants then inspected, IE Bulletin No. 83-02 was issued in March 1983, to extend the inspection requirements to all other BWRs. In addition, it required that a larger sample of welds be inspected, with provisions for further increase in the inspection sample if cracks were found.

These inspections continued to indicate that cracking in most BWRs was extensive, and many required major repairs by weld overlay reinforcement. Because the reports indicated that many welds required repair to meet ASME Code requirements for further operation, the NRC determined that five plants still to be inspected should be ordered to inspect as soon as practicable, and should inspect essentially 100% of all welds in the recirculation system and connecting systems that were considered to be susceptible to IGSCC.

3.2 INSPECTION RESULTS

The results of all of the BWR inspections conducted under IEB 82-03 and 83-02 are summarized in Table 3.1. This covers the first round of inspections, and is complete as of March 1984. The inspection results show wide variations in extent of cracking reported. Some plants, Oyster Creek, Duane Arnold, Fitzpatrick, and Millstone 1, for example, reported very little, if any, cracking. Other plants such as Dresden 3, Cooper, Hatch 2, Vermont Yankee, and Browns Ferry 1 reported very extensive cracking. At this time no obvious reasons for this wide variation have been determined.

In addition to the welds listed in Table 3.1, a number of licensees inspected welds in systems with smaller-diameter piping. Significant cracking was also found in these systems (RWCU, Core Spray, etc.), as expected. Augumented inspections of these systems were already required by NUREG-0313, Rev. 1, and are therefore not of primary interest in this report.

As a direct result of this inspection program, utilities are planning to take positive actions to preclude or minimize IGSCC in the future. Several utilities are in the process of replacing susceptible piping with Type 316NG stainless steel, which is highly resistant to sensitization and IGSCC. Monticello, Hatch 2, and Pilgrim are currently undergoing such replacement. Several other utilities are making positive plans, including procuring replacement material.

The NRC has permitted plants to operate for one fuel cycle with weld-overlay-reinforced cracked welds, or with welds with minor cracking that will

TABLE 3.1 Summary of All Inspection Findings on Large Piping in All Operating BWRs Inspected According to IEB 82-03 and 83-02

Plants	Extent of Inspection (% of welds inspected) Recirc.	RHR	Inspection Results (No. of cracked welds) Recirc.	RHR	No. of Welds Overlay Repaired
Big Rock Point	20% (11/59)	---	0	--	0
Browns Ferry 1	98% (103/105)	90% (36/40)	33	14	42
Browns Ferry 2	27% (25/91)	28% (9/32)	2	0	0
Browns Ferry 3	98% (103/105)	28% (9/32)	0	0	0
Brunswick 1	25% (29/115)	75% (3/4)	3	0	3
Brunswick 2	100% (102/102)	100% (5/5)	15	1	8
Cooper	100% (108/108)	100% (7/7)	20	0	13
Dresden 2	47% (47/101)	10% (4/40)	10	0	7
Dresden 3	100% (115/115)	90% (45/50)	53*	11*	61
Duane Arnold	42% (49/117)	40% (2/5)	0	0	0
FitzPatrick	47% (49/106)	45% (5/11)	1	0	0
Hatch 1	47% (47/100)	100% (11/11)	5	2	6
Hatch 2	94% (97/103)	100% (11/11)	36	3	27
Millstone 1	11% (11/100)	0% (0/46)	0	0	0
Monticello	100% (106/106)	78% (18/23)	6	0	6
Nine Mile Pt. 1	82% (62/76)	---	53	0	0
Oyster Creek	39% (31/80)	---	0	0	0
Peach Bottom 2	100% (91/91)	91% (32/35)	19	7	21
Peach Bottom 3	91% (77/85)	92% (35/38)	10	5	15
Pilgrim 1**					
Quad Cities 1	8% (9/110)	20% (9/44)	0	0	0
Quad Cities 2	100% (106/106)	90% (45/50)	20	2	9
Vermont Yankee	66% (58/88)	7% (2/30)	33	1	22

*It should be noted that 18 welds originally reported to be cracked were later reevaluated and determined not to be cracked, so are not included in these totals.

**After inspecting approximately 7 welds, and finding cracks in 4 of them, the utility decided to replace the piping with Type 316NG, so has not completed the examination.

remain well within ASME Code limits for one fuel cycle. Other utilities, with less extensive cracking, are considering alternative measures. Some utilities have performed IHSI on welds with minor cracking to preclude further crack extension. Decisions regarding the acceptability of further operation under the above conditions are needed.

3.3 FOREIGN EXPERIENCE

A special meeting of the Committee on the Safety of Nuclear Installation (CSNI) was held at the NRC's request in February 1984. The purpose of this meeting was to review the experience of IGSCC among the various countries with operating BWRs, and to discuss technical and licensing aspects of the problems. A summary of the incidents reported by the representatives follows.

3.3.1 Japan

A total of 43 welds at six plants were found to have IGSCC. Most of the cracking was discovered before 1978. Thirteen of the 43 cracked welds were detected by leakage; the others (30) were detected by NDE. Thirty-two cases of IGSCC were in pipes 4 inches or larger in diameter (up to 22 inches); the remainder were in the smaller-diameter piping. No cracks have been found since 1982, probably because of extensive mitigating actions, including corrosion resistant cladding (CRC), solution heat treatment (SHT), IHSI, or replacement with 304NG or 316NG. In all known cases, the cracked pipe has been replaced. In plants not yet operating, recirculation piping has been or is being replaced with material such as 316NG. In operating plants, replacement with 316NG or similar materials has not been completed but has been done selectively.

The water chemistry of Japanese plants, particularly with regard to residual ions contributing to conductivity, is about one order of magnitude better than that in U. S. plants, which probably contributes to the lower incidence of IGSCC in the older sections of Types 304 or 316 SS Deaeration during startup markedly reduces the level of dissolved oxygen prior to ascent to power (e.g., from approximately 3.0 ppm to approximately 0.1 ppm).

3.3.2 Spain

During the 1983 outage, a large sample of welds was inspected at the Santa Maria De Garona Plant. In the recirculation system, 66 out of a total of 154 welds were inspected and cracks were reported in 15. Two additional cracks were found, one in the LPCI system, and one in the CRD return line. Actions taken were similar to those being taken in the United States. Four welds were repaired by weld overlay, two were repaired by other means, and the remaining eleven were evaluated as acceptable for further operation.

3.3.3 Germany

In the newer German BWR's, Type 347NG stainless steel is used for piping less than 12-in.-diameter. This is an alloy that is stabilized against sensitization; therefore IGSCC would not be expected. Larger diameter piping is carbon or low alloy steel clad with Type 347 weld metal. The main recirculation piping in Gundremmingen Unit A (a dual-cycle BWR) was made of Type 304 stainless steel. Safe ends were furnace sensitized. In 1978, cracks were found in the 24-in.-diameter safe ends and piping at the safe-end-to-pipe welds. This is the only case of IGSCC in stainless steel reported in Germany. This reactor has now been decommissioned.

3.3.4 Switzerland

One crack was found and identified as IGSCC in the recirculation bypass line at the Muehleberg plant. The bypass line was removed and capped. Recent inspections, in 1982 and 1983, disclosed a cracklike indication in a weld to the main recirculation discharge valve. Although the reported inspections disclosed no growth during this period, the utility plans to replace this section of pipe in 1984.

3.3.5 Sweden

Although most of the piping in Swedish BWRs is clad ferritic steel, several cases of IGSCC have been found in Type 304 SS in Ringhals Unit 1 and Oskarshamn Unit 1. Most cases were in relatively small-diameter piping, and were attributed to combinations of fabrication problems. These were of concern, because it was believed that the relatively low carbon level (less than 0.05%) required in Sweden would minimize IGSCC.

In 1982, leaks were noticed in the 4-in.-diameter control rod drive hydraulic system. An ultrasonic examination was conducted on all 4-in.-diameter piping, and 39 indications were found, of which six had to be repaired. Because the piping involved had a relatively high carbon content, (near the 0.05% limit), it was all replaced in 1983 with low-carbon material. Subsequent examinations showed that most of the indications reported were not cracks.

One small leak was found in a bypass line in Barseback 1 during 1983; again, the problem was attributed to poor fabrication processes.

In summary, there have been only about 20 incidents of IGSCC in Swedish plants, primarily in 4-in.-diameter piping welds. The Swedes are actively pursuing hydrogen water chemistry as a remedial action to prevent future incidents of cracking.

3.3.6 Italy

The only case of IGSCC reported in Italy occurred at Garigliano in 1978. A safe-end weld was removed, and destructive examination disclosed IGSCC. No IGSCC has been found to date in the Caorso plant, but there are plans to perform an augmented inspection using improved methods during 1984.

3.4 CONCLUSIONS

- Although significant IGSCC was found in the United States in the past, no cracking in large pipes was reported until recently. As a result of extensive inspections performed during the past two years, IGSCC has now been reported in 19 out of 23 plants. All other countries have experienced IGSCC in piping made of Type 304 stainless steel, including large-diameter Type 304 stainless steel piping, in those plants that have it.

- In all other countries the preferred long-term solution is to replace piping made of Type 304 stainless steel with carbon steel or one of the low-carbon or stabilized grades of stainless steel. Type 316NG is considered the preferred choice, except in Germany, where they prefer their special nuclear grade of Type 347.

- Interim or short-term fixes in other countries follow the same approaches as have been used in the United States, and include IHSI, weld overlay reinforcement, welded "clam shell" reinforcement, last-pass heat sink welding, and interim operation with unrepaired small cracks. It should be noted that IHSI on uncracked Type 304 stainless steel pipe is considered to be a permanent fix in Japan.

- Hydrogen water chemistry is being seriously considered by several countries as a desirable adjunct to other solutions.

4.0 NONDESTRUCTIVE EVALUATION OF PIPING WELDS

4.1 INTRODUCTION

Early detection of cracking in primary piping systems is important for economic as well as safety considerations. Ultrasonic inspection is one of the major technologies for early detection of pipe cracks.

Recent field experience and round robin tests have demonstrated a need for improvement, as well as some possible mechanisms for improving the reliability of crack detection, interpretation, and characterization. In addition, the NRC and industry have supported extensive research efforts aimed at quantifying the capabilities of nondestructive evaluation techniques.

This chapter will discuss:

- Current Ultrasonic Examination Requirements
- Technical Problems of UT Inspection
- Actual Field Experience before and after the issuance of IE Bulletins 82-03 and 83-02
- UT Round Robin and Laboratory Experience
- Recent Improvements in Ultrasonic Inspection
- Personnel, Equipment, and Procedure Qualification

- Foreign Experience
- Leak Testing Requirements and Techniques
- Conclusions and Recommendations

4.2 CURRENT ULTRASONIC EXAMINATION REQUIREMENTS

4.2.1 Applicable Requirements

The preservice and inservice examination requirements for the piping systems in nuclear power plants are set forth in the Code of Federal Regulations, Part 50, Section 50.55a, which references the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Section XI is published every three years, with semiannual Addenda.

In determining which edition and addenda of Section XI are applicable, the following rules are used. (However, utilities may choose to update to later editions of the Code.)

- The edition of Section XI to be used for a plant's preservice inspection program (PSI) is determined by the date of issuance of the plant's construction permit.
- The edition of Section XI to be used for a plant's inservice inspection program is determined by the date of issuance of the plant's operating license.

During the intervening time -- the duration of construction -- Section XI has typically undergone significant changes. These fundamental differences in examination criteria undermine the major purpose of the PSI which is to provide an NDE baseline measurement of the examined components, for comparison with the findings of subsequent inservice inspections.

4.2.2 Example of Requirement Changes

An example of the far-reaching effects of changing Code editions is illustrated in the change in ultrasonic calibration reflectors between the 1974 edition of Section XI (Summer 1975 Addenda) and the 1977 edition of Section XI (Summer 1978 Addenda).

The 1974 edition requires that calibration be performed using side-drilled holes, located at either 1/2T, or at 1/4, 1/2, and 3/4T (depending on wall thickness). The 1977 and later editions require calibration on end-milled notches that are 10% of the wall thickness in depth. These notches are placed on the inner and outer surfaces of the block.

The impact of the change in calibration reflectors between the 1974 edition of Section XI (Summer 1975 Addenda) and the 1977 edition of Section XI (Summer 1978 Addenda) was investigated by PNL under two NRC programs (4.1, 4.2). Data from these studies consisted of approximately 540 measurements on 34 piping calibration standards. The standards were from a boiling water reactor (BWR) currently under construction and ranged in wall thickness from 0.237 to 2.343 inches. Twenty-six were carbon steel and eight were stainless steel. The samples contained both side-drilled holes and notches in accordance with Code requirements.

Measurements were performed by establishing a distance amplitude curve (DAC) using the side-drilled holes and then measuring the response of the notch relative to this DAC curve. The notches produced responses that exceeded the response of the side-drilled holes. A calibration performed using notches will, therefore, be less sensitive than a side-drilled hole calibration. This reduction in sensitivity depends on wall thickness, ranging from -6 dB (a factor of 2) at 0.4 in. to -16 dB (a factor of 6.3) at 2.4-in. wall thickness.

Theoretical calculations were performed to assure that these experimental results were reasonable. The theoretical calculations were in close agreement with the measured results.

Thus, the results of this study show that preservice examination performed to the requirements of the 1974 edition of Section XI may be significantly more sensitive than inservice examinations performed to the 1977 edition of Section XI. Consequently, preservice and inservice examinations may have different reliabilities (4.1, 4.2).

4.2.3 Comments on Standard Review Plans and Generic Safety Issues

Sections 5.2.4 and 6.6 of the U.S. NRC Standard Review Plan (NUREG-0800) outline the review and compliance criteria for inservice inspection of ASME Class 1, 2, and 3 components. The Standard Review Plan criteria referenced in Sections 5.2.4 and 6.6 are the requirements of Section XI of the ASME Code. As discussed in the preceding paragraph, changes in Code requirements can cause significant variation in inspection reliability. Therefore, review criteria in Standard Review Plans may be inadequate because Code requirements are inadequate.

Generic Safety Issue A-42, "Pipe Cracking in Boiling Water Reactors," was addressed by NUREG-0313, Revision 1. The report requires, in part, that all applicants/licensees of BWR plants provide a program for replacement of service-sensitive lines. If design criteria for replacement of lines includes consideration for ultrasonic inspection, improvement in subsequent ultrasonic examination reliability would be possible.

Generic Safety Issue A-14, "Flaw Detection," has been dropped. The rationale for this is that flaw detection is not by itself a safety issue. Flaw detection is important on a problem-specific basis, i.e., pressurized thermal shock. Research and development work is occurring through the U. S. NRC, Office of Nuclear Regulatory Research, and the industry to improve flaw detection reliability. The Task Group on Pipe Cracking supports these efforts and supports timely completion of the work for resolution.

4.2.4 Comments on Current ASME Code Requirements and Suggested Improvements

Round robin tests (4.5) and other studies (4.3) have shown inadequacies in ASME Code requirements. Research activities in the Office of Nuclear Regulatory Research and the industry and field experience have identified methods and means to improve the Code. The following list details areas where inspection reliability can be improved and inspection variability can be reduced.

- Code Case N-335 - This Code case should be made mandatory for all required inspection.

- Calibration Blocks - For austenitic stainless steels, the calibration blocks should contain welds, and the calibration reflectors should be located either in the weld or on both sides of the weld. In addition, the calibration block and pipe should have the same nominal microstructure.

- Calibration Reflectors - When notches are used as calibration reflectors, compensation should be required for the sensitivity differences between notches and side-drilled holes.

- Search Units - In addition to the currently required 45° shear wave examination for welds, an additional 60° shear wave examination should be required. The potential advantages of the refracted longitudinal wave technique for dissimilar metal welds, weld metal, and far-side inspections should be recognized, and this technique should be required as a supplementary examination.

- Beam Spread - If beam spread corrections are made as permitted by the ASME Code, flat calibration reflectors should be used for the corrections to crack and lack-of-fusion-type indications.

- Angled Defects - A skewed scan should be required to detect defects oriented other than parallel or perpendicular to the weld.

- Crack Length Sizing - The 50% DAC method of crack length sizing should be revised to require that end points of a flaw be determined by loss of signal amplitude to the background noise level.

4.3 TECHNICAL PROBLEMS OF ULTRASONIC INSPECTION

Ultrasonic inspection of piping is hindered by a number of conditions that are encountered in the field. This section considers those conditions which have the largest adverse impact on the inspection.

4.3.1 Component Access, Geometry, and Environment

The inside surface contour of a pipe weldment affects the ultrasonic examination. The presence of a number of conditions such as counterbores, inner surface cladding, mismatch, weld root drop-through, or suck-back can produce nonrelevant ultrasonic indications that can be misinterpreted as being caused by flaws, sometimes resulting in unnecessary repairs. Conversely, since at least one of these conditions is usually present, the UT operator may mistakenly attribute a flaw indication to one of these ID geometric reflectors.

Usually IGSC cracks occur in the weld heat-affected zone (HAZ) near the weld root. Since the root of a stainless steel weld usually returns a signal, it can be difficult for the UT operator to distinguish between the root indication and an adjacent crack indication. A crack indication may be mistaken for an irrelevant root indication, or a root indication may be misinterpreted as being from a crack. An unambiguous discrimination can sometimes be achieved by making careful measurements of search unit position and pulse transit time for plotting purposes, but these measurements increase the time spent in the radiation area, and are subject to errors caused by beam redirection (bending) effects.

Ultrasonic inspection is further complicated by inadequate access for examination. Usually pipe joint design and installation are performed without consideration for adequate access for UT weld inspection. Access is often available from only one side of the weld, and may be further limited by the presence of pipe supports or other restraints. Poor outer surface finish and/or excessive weld crown contour can also reduce the effectiveness of UT examination.

Ultrasonic inspection is most effective when applied in a deliberate and careful manner. In the environment of a nuclear power plant, however, this is often not possible. The inspections may be performed under conditions of great discomfort due to high temperature and humidity, exacerbated by the necessity for wearing respiratory equipment and multiple layers of protective clothing. Accurate manipulation of the search unit is necessary with simultaneous visual contact with the UT instrument, but since the pipe runs are not designed for inspection, the UT operator must often assume very awkward positions in order to reach the examination areas. Finally, health physics considerations frequently require that the examination be performed in a very short time.

4.3.2 Effect of Material Properties on Inspection

The acoustic properties of the material affect the propagation of ultrasound. Sound beams encountering anisotropic grain structures undergo reflections and refractions that do not occur in grain structures of isotropic materials. The grain size may vary from point to point in the pipe base material, and will vary between the pipe base metal and the HAZ. The weld metal will also have different grain size and structure than either the HAZ or the pipe base metal. These variations in grain size and grain structure tend to increase the attenuation of the beam and also return irrelevant reflections that may complicate the instrument screen presentation. These effects are most severe in cast stainless steel materials and stainless steel welds.

The cumulative effects of refraction at many grain boundaries can cause the sound beam to travel in a curved path instead of the expected straight line, resulting in mislocation of indications and sometimes in failure to insonify* parts of the required inspection volume. This is particularly a problem in attempting to inspect the far side of a stainless steel weld by directing the sound beam through the weld metal. When applying the conventional shear wave examination required by the Code, the sound beam may be guided by the weld metal grain structure to the weld root, and the far-side HAZ may not be inspected.

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*The definition of insonify is to provide the proper sound field distribution throughout the volume of material undergoing examination.

Variable UT attenuation in the base metal causes the sensitivity of the examination to vary. Methods of measuring and correcting for these variations exist, but they significantly increase the inspection time and radiation exposure.

Many cracked welds have recently received overlays of stainless steel weld metal on the outer surface, as a means to achieve reinforcement and stress reversal for a temporary remedy pending pipe replacement. These weld overlays present the same inspection problems experienced in cast stainless steel. Attenuation in the overlay will be high and may also vary with location, the beam angle may be unpredictable, and a high UT noise level will probably be present. Detection of new cracking and/or monitoring known cracks below the overlay may be unreliable when using manual (conventional) UT equipment.

4.4 FIELD EXPERIENCE

4.4.1 State of UT Practice in ISI Before Mid-1982

Before the issuance of IE Bulletin (IEB) 82-03 in October 1982, UT procedures reflected the requirements of two documents: ASME Code Section XI, and NUREG-0313, Revision 1. Section XI defined the basic minimum requirements for examining BWR piping. NUREG-0313, Revision 1 required augmentation of the inspection sample and inspection frequency to include a larger sample and more frequent inspection than specified by Section XI. This document also required that UT examinations be performed in accordance with Section XI, but noted that the Code minimum requirements may be inadequate for IGSCC detection.

Hence, most utility and ISI vendor UT procedures designed for NUREG-0313, Revision 1 applications are enhanced beyond the Code minimum requirements. The most common enhancements are added sensitivity, the requirement that search units be optimized for IGSCC detection, and inclusion of scanning motions appropriate for detection of IGSCC that is neither parallel nor perpendicular to the weld.

4.4.2 IE Bulletins 82-03 and 83-02

As discussed in Section 3.1, leaking IGSC cracks were visually detected, in March 1982, at two recirculation system safe-end welds in Nine Mile Point Unit 1. The welds had been ultrasonically inspected nine months earlier, but no reportable indications were discovered. After the leakage, axially-oriented inner surface cracking was confirmed using a modified UT procedure. Additional examinations of large-diameter recirculation welds disclosed inner surface cracking adjacent to a large number of the welds.

As a result of the experience at Nine Mile Point, IE Bulletins (IEB) 82-03 and 83-02 were issued in late 1982 and early 1983. IEB 82-03 was applicable to nine BWR plants (operated by eight different utilities), and required that the UT methodology and detection capability for use on recirculation system welds be demonstrated on service-induced cracked samples. The demonstration required by IEB 82-03 showed a tendency toward both severe overcalling, and severe undercalling, since some procedures and equipment were inadequate for the task. The major benefit of IEB 82-03 was probably the demonstration to the NRC, the licensees, and the ISI vendors of the overall ineffectiveness of the UT/ISI inspections that were conducted prior to late 1982.

The issuance of IEB 83-02 had an even greater impact since it applied to 14 additional BWRs, and required that each field UT/ISI team demonstrate its UT process using a minimum of 80 linear inches of service-induced crack samples. An 80% detection rate was required, and a penalty was imposed for false calls. The general requirements included a six-hour time limit, and NRC review of the proposed procedure. IEB 83-02 also specified an augmented ISI sampling plan which included (as a minimum) ten welds in recirculation pipes larger than 20 inch diameter, ten welds in recirculation risers and safe ends, and two sweepolet-to-manifold welds near end caps. If cracks were found, additional sampling per IWB-2450 was required.

The results from the performance demonstration data of IE Bulletin 83-02 were:

- More meaningful demonstrations (relative to IEB 82-03) were achieved owing to the requirements for a time limit, a penalty for false calls, specified size and makeup of the UT team, additional specimens, specific pass/fail criteria, and better uniformity between NRC Regions.

- The overall detection capability of the UT process (personnel, equipment, and procedures) for BWRs was definitely enhanced as a result of passing IEB 83-02. The basis for this statement is the dramatic increase in the number of field indications detected and classified as IGSCC per inspection. A fully definitive evaluation of these results is not yet possible pending completion of the correlation of UT results with further examination of pipe welds removed from the field.

- Since each team's success was the result of their combined efforts, individual team member performance could not be evaluated.

- The limited number of specimens and cracks does not permit a statistically significant assessment of performance.

A psychological factor is also involved. Before the Nine Mile Point experience, an operator claiming to have found a crack in a large recirculation pipe could expect to meet skeptical resistance, because of the general perception that these pipes were not susceptible to IGSCC.

Because of field experience, the results from the bulletin performance demonstrations and the rather discouraging results from round robin studies of UT crack detection capability, the EPRI/NDE Center initiated development and implementation of a series of one-week training courses on IGSC crack detection using test specimens removed from cracked Nine Mile Point piping (4.3). The course outline includes UT theory, IGSCC morphology, a generic procedure, laboratory exercises to improve operator proficiency, and a three-part final examination. Each student's performance is reported to his employer for certification use, although the NDE Center does not certify successful graduates.

As of early January 1984 this course was being offered at least monthly, with about 15 students per class. The passing rate for students has been about 50%, and passage of this course has been considered as one acceptable way to meet the requirements of IEB 83-02. The major benefit of such courses is that the same information is being distributed to all who need it; hence, some element of industry-wide standardization is occurring. However, the number of specimens being used during the final, practical examination is insufficient to provide high statistical confidence levels. On a statistical basis alone, the performance demonstrations resulting from IEB 83-02 may be analyzed from two viewpoints: as a measure of individual inspector capability, or as a measure of the average capability of the inspector population. A probability of detection of at least 42% at a 90% confidence level is calculated for an individual inspector capability using classical statistical methods (binary) for a 4 out of 5 test. Using Bayesian statistics and assuming a 20% false call probability during the demonstration tests, the average capability of the inspector population (for those passing the demonstration test) is calculated at a level of 65% probability of detection. However, the Nine Mile Point specimens in IEB 83-02 performance demonstrations may not be typical of IGSCC conditions in other BWR plants and the testing environment may not be a reasonable approximation to an in-field environment and therefore complicates the extrapolation of the test results to field performance.

Even with the foregoing, it is quite evident that IE Bulletins 82-03 and 83-02, and the resulting IGSC crack detection demonstration and training courses at the EPRI/NDE Center, have produced both positive and tangible results. The augmented-type inspections conducted since these have been in effect have generally resulted in much higher reporting rates for IGSCC in the primary piping systems of BWRs, and a corresponding general increase in the confidence that can be placed in these examinations. To maintain this level of proficiency, there should be a requirement that plant examinations be conducted using the same equipment and procedures used to pass performance demonstration tests to eliminate the possibility that a successful inspector could still be an ineffective operator by using equipment or procedures that are inherently not capable of detecting IGSCC.

4.4.3 Crack Sizing

In late 1983, the EPRI/NDE Center initiated a "crash" program to develop and implement an additional one-week training course on UT crack sizing. This course has also been structured in the "module" format, and each of several sizing techniques will involve modules and theory and procedures, plus some laboratory exercise time. Written and practical examinations will be given to evaluate student performance. The techniques to be covered will include: amplitude drop, sound trapping (full V-path), various crack-tip detection methods such as shear and longitudinal waves at 1/2 and full V-paths, and high angle longitudinal waves (also called creeping waves).

Because UT crack sizing often produces questionable results, even under laboratory conditions, the actual value of this crack sizing course remains to be seen. Certainly, some direct benefits will accrue since all of the students will develop a greater appreciation for the current difficulties. However, it should be recognized that the availability of a course and its graduates may not necessarily result in a routine and reliable process for crack sizing.

4.5 ROUND ROBIN EXPERIENCE

Two UT round robins have been conducted in response to piping problems. A summary of the results from each follows in the next two subsections.

4.5.1 PNL Round Robin

A round robin test of pipe inspection reliability was initiated in 1981 at the Pacific Northwest Laboratory (PNL) under NRC sponsorship. The objective was to measure the crack detection and sizing reliability of inspection teams from commercial ISI vendors (4.4).

Three-man inspection teams from six commercial ISI vendors participated in the Pipe Inspection Round Robin. Each team consisted of Level I, II, and III examiners, selected to represent neither the best nor the worst levels of ISI experience available.

Each team spent three weeks at PNL, working eleven hours per day six days per week to complete a carefully designed test matrix of 253 separate inspections per team. A variety of inspection conditions were simulated, but in almost all cases the specimens were masked to permit inspection access from only one side of the weld.

The results and conclusions drawn from this pipe inspection round robin test were as follows:

- UT detection of cracks in clad ferritic main coolant pipe can be 100% effective, if adequate sensitivity is used per ASME Code Case N-335. Section XI minimum sensitivity (50% of the notch amplitude) is not adequate.

- Detection of cracks in clad ferritic pipe is almost equally effective with and without weld metal in the sound path.

- UT detection of cracks in centrifugally cast stainless steel (for PWRs) is ineffective using the conventional manual techniques currently applied in the field. The false call rate was almost identical to the probability of detection and correct interpretation (PODI) rate.

- Section XI minimum requirements do not provide effective inspection of wrought stainless steel pipe welds. Increased sensitivity and selection of optimized search units improves detection reliability. For both IGSC and thermal fatigue cracks in stainless steel, the six teams achieved an average PODI of 50-60% when using their own procedures for cracks 15% throughwall or greater.

- When the sound beam must pass through the butt weld in wrought stainless pipe, UT inspection using current field techniques is ineffective.

- Variability in crack detection reliability is significant from operator to operator, even when identical equipment and procedures, are used.

- Crack detection reliability in UT inspection of stainless steel pipe welds should be qualified by test.

- Crack length measurements were in general quite good. There was a trend to oversize very small cracks and to undersize very long cracks by small amounts. A conservative approach would be to record length based on signal reduction to the background noise level.

- Crack depth measurements by all the teams using the Code-advocated method of amplitdue drop was totally ineffective.

4.5.2 EPRI Sizing Study

EPRI conducted a round robin test of IGSC crack depth measurement capability in 1983 (4.3). The objective of the exercise was to generate information necessary to assess the current state of practice of ultrasonic depth measurements of IGSCC found in the piping of some BWRs. Seventeen teams from ISI vendors, utilities, research organizations, NRC, and one foreign utility participated. Thirteen IGSC cracks and three EDM notches were included in the test. Two of the cracks were removed from the Nine Mile Point BWR; the rest were laboratory-cracked pipe from PNL and IHI in Japan. All of the deep reflectors were either notches or wide open laboratory-induced IGSCC. After the UT measurements were finished, nine of the specimens were destructively evaluated to determine the true crack depths.

The results showed a general tendency to oversize the cracks that were less than 20% throughwall and to undersize those deeper than 20%. However some teams definitely performed better than others (4.3). Two advanced UT techniques were also included in the round robin. Their effectiveness was similar to that of the better manual teams.

The EPRI report contained the following conclusions (direct quote):

"The results show the following:

- The number of teams doing an adequate job is much lower than anticipated.
- The range of performance from best to worst is very large.
- The advanced techniques provide verification that the crack-tip diffraction sizing approach is the most viable method.
- The influence of the ultrasonic measurement uncertainty to the flaw evaluation results must be assessed on case-by-case basis.
- Corrective actions are needed."

Analysis of the EPRI sizing study results in the following additional observations:

- The most important conclusion of the EPRI sizing study was that none of the teams in the study could size accurately. Even though some of the teams showed a significant relation between measured and true crack size, the scatter of the measured data is so large that crack depths could not be determined accurately; at best ± 1/3t.

- A relatively small number of nonrepresentative reflectors were used during the EPRI sizing study. This reduces the accuracy with which the team's performance can be determined, particularly for deep cracks. Furthermore, the thick-walled specimens contained only shallow cracks, so the regression results were largely determined from thin-walled pipe specimens which contained a broader range of crack sizes.

- The use of EDM notches and wide cracks could tend to aid crack sizing performance, particularly for techniques relying on crack-tip signals. Since most field cracks have been found by destructive analysis to be tight, basing sizing performance on EDM notches and wide cracks is nonconservative.

4.6 RECENT IMPROVEMENTS IN ULTRASONIC INSPECTION

4.6.1 Qualification of Personnel and Procedures

The major impact of IE Bulletins 82-03 and 83-02 has been in demonstrating inspection capability. While the concept of demonstrating inspection capability is not new, the IE Bulletins marked the first effort to demonstrate inspection capability on a national scale. The results of the demonstrations required by IEB 83-02 showed a dramatic increase in crack reporting during in-service inspection. This increase in crack reporting has not been without cost. The results of inservice examinations at Dresden Unit 3 and other plants has confirmed several cases of false calling.

4.6.2 Automated Ultrasonic Systems

A variety of automated UT systems have been manufactured. Few of these systems are commercially available. Data appropriate for comparing automated system performance to manual UT performance are rare.

The inspection functions which have been automated vary from system to system. For purposes of comparing one automated system to another, the inspection functions may be broken out into data collection, data recordings, and data interpretation.

Automated data collection by mechanical scanners improves the accuracy and repeatability of indication location and facilitates comparison of the data from successive ISIs. The use of mechanical scanners reduces the need for highly skilled UT operators to remain in radiation zones for general data collection and thereby, expanding the capability for taking whatever limited special manual data that an operator may choose.

In some instances, mechanical scanners may be ineffective or even unusable. Current design limitations generally restrict scanners from use on complex weldment geometries such as those found in branch connections, sweepolets, etc. Some automated scanners are not capable of skewing the search unit from side to side, which is necessary for detecting IGSC cracks oriented at an angle to the weld, and for optimizing the UT response from other obliquely oriented planar reflectors. Physical access for mounting some scanner designs may be precluded by the presence of pipe supports, restraints, branch connections, etc.

Automatic techniques can record information on signal properties and physical coordinates many orders of magnitude more quickly than manual techniques. More complete information can be recorded while still reducing the UT operators' radiation exposure times. Automatically recorded data are more accurate and less error-prone than manually recorded data. Automated techniques tend to record raw, unreduced data, thus allowing the possibility of reprocessing archived raw data using new interpretation schemes as they are invented.

While data interpretation can be broken out logically as an inspection function, it should be kept in mind that it requires compatible data records and in some cases data collection using scanners with exacting specifications. Thus, the systems that automatically interpret data also tend to collect and record data automatically.

Automated interpretation eliminates the occurrence of random errors and nonuniform interpretations caused by humans. Depending on the interpretation algorithm used and its implementation, a speed advantage over human interpretation may result; on the other hand, some current algorithm implementations are not compatible with real-time operations.

Automated data interpretation involves manipulation of the data, either by directly extracting waveform features, or by first using the waveforms in an imaging scheme and then extracting image features. In either case, these features are correlated with features previously associated with particular types, sizes, shapes, and orientations of defects.

Southwest Research Institute developed a computer program called FLAWSORT that interprets waveforms to be either from cracks or from geometry on the basis of a preset algorithm. Adaptronics sells an Adaptive Learning Network (ALN) that performs similarly, on the basis of an algorithm that it generates while "learning" the difference between data it is told represent cracks and data it is told represent geometry.

Several systems perform imaging: the Synthetic Aperture Focusing Technique (SAFT) developed under NRC sponsorship at the University of Michigan, Southwest Research Institute and currently at Pacific Northwest Laboratory; the Advanced Ultrasonic Testing System (AUTS) at EG&G Idaho; the Projection Image Scanning technique (P-Scan) developed by the Danish Welding Institute; the Ultrasonic Data Recording and Processing System (UDRPS) at Dynacon Systems in California; the Ultra-Image III developed by General Dynamics; the ALN 4060 developed by Adaptronics; and acoustic holography as practiced at several organizations. Most of these imaging systems have not yet implemented fully automated image interpretation.

A research program sponsored by the NRC is currently evaluating the merits of several of the above advanced UT systems. A report from the program is scheduled for June 1984 publication. Preliminary data are encouraging particularly for some of the imaging systems. Difficulties with the mechanical scanners are a major limitation for application of the systems in the field.

4.7 PERSONNEL AND PROCEDURE QUALIFICATION

Although a general upgrading of qualification requirements does not automatically assure more effective field inspections, it does increase the intrinsic capability for a reliable inspection and intensifies the awareness of the field-inspection personnel to the significance of their work.

4.7.1 Qualification Document

Improvements in the detectability of intergranular stress-corrosion cracking (IGSCC) on reactor piping systems by the ultrasonic inservice inspection teams following implementation of IE Bulletins 82-03 and 83-02 showed that upgraded qualification requirements can provide definite benefits. However, further improvements in overall effectiveness of the ultrasonic inservice inspection (UT/ISI) processes are considered to be both necessary and achievable. Toward this end, the NRC Office of Nuclear Regulatory Research is developing a document that specifies more rigorous qualification requirements than those now in effect.

The scope of the document will include qualification requirements and criteria for the critical elements of a UT/ISI system; namely, personnel, equipment, and procedures. The document will specify both general and specific requirements and criteria for the qualification and requalification of UT personnel, equipment, and procedures.

The most significant features of the qualification approach specified in this document are:

1. A statistically significant performance qualification demonstration is required.

2. The UT personnel, equipment, and procedures are all considered to be critical inspection elements, and all three are involved in the qualification process.

3. Classroom training is required prior to initial qualification, and annually thereafter.

4. Guidelines are provided so that users of the document can generate application supplements.

The performance qualification demonstrations are intended to assure minimum levels of inspection capability. Limited qualification will be possible; for example, a UT operator may be qualified to interpret indications, but not to characterize those indications which are interpreted to be flaws.

This document will specify requirements and processes for conducting "blind test" performance demonstrations to quantify the flaw detection probability and characterization accuracy for candidate UT/ISI systems. The document will also specify the flaw detection probability and characterization accuracy that must be demonstrated as a qualification for performing UT/ISI.

Although the implementation of such a program is not a trivial consideration, little emphasis has been placed on this aspect to date. Rather, the initial approach has been to focus most of the effort on developing and refining the technical aspects of the program. Once the program has been defined and refined, the emphasis will be shifted towards addressing the implementation aspects.

4.7.2 Ad Hoc Utility Committee Document (NUC-MR-1A)

In 1982, the Ad Hoc Committee for Development of Qualification Requirements for Nuclear Utility Examination Personnel was established for the purpose of developing minimum requirements for qualifying NDE personnel, and recommendations for implementing these minimum requirements. This group submitted a proposed minimum requirements document (NUR-MR-1A) to the NRC in September 1983. The Ad Hoc Committee document (NUC-MR-1A) represents both an expansion of the "recommended practice" document (SNT-TC-1A) that is published by the American Society for Nondestructive Testing, and an effort to establish a set of "minimum requirments" in lieu of the general "guidelines" that are contained in SNT-TC-1A.

The result, however, is still a document that will require each user (utility or ISI contractor) to develop an individualized "Written Practice" to specify how that employer will comply with the "minimum requirements" contained in NUC-MR-1A. This approach could be viewed as a basic weakness of the NUC-MR-1A document since users may be more inclined to tailor their written practices to their current programs, rather than to the spirit and intent of NUC-MR-1A. Although the Ad Hoc utility committee document did increase some of the basic guidelines from SNT-TC-1A with respect to experience, number of exam questions, etc., many of the additional requirements in NUC-MR-1A can be regarded as rather superficial. The document, NUC-MR-1A, does contain an added section (beyond SNT-TC-1A) entitled "Qualification for Special Applications," wherein the need for special qualifications are recognized. However, this section (consisting of three brief paragraphs) simply states that "When circumstances require the demonstration of additional examination capabilities, the Level III examiner shall determine the need for qualification of personnel for such special applications." This section goes on to say that special techniques or equipment or problems may require special procedures, examinations, and training. Furthermore, such special qualifications must be documented.

If, and when, this document is implemented it is expected that the status quo will not change because most employer's written practices will require only minor changes and will therefore, be inadequate.

4.7.3 Proposed Appendix to ASME Section XI

The Section XI Working Group on Nondestructive Examination (WGNDE) is developing a procedure and personnel qualification requirements Code Case for Section XI. The proposed approach is patterned after qualification processes performed under IEB 83-02.

The document defines the requirements for qualification of ultrasonic examination procedures and personnel for detection and sizing of IGSCC. The qualification requirements apply to personnel who record, interpret, characterize, and size IGSCC flaws.

The document requires that both procedures and personnel be qualified on five IGSC cracks and provide requirements for the other aspects of the qualification test matrix. The following requalification requirements are being proposed:

- When an examiner has not found confirmed IGSCC flaws or participated in a blind test on a single approved test specimen for a period of six months.

- When there is specific reason to question his ability.

The document provides the following list of essential variables.

- Instrument or system - make and model.

- Search unit - size, type, angle, frequency, manufacturer's model.

- Methods and criteria for establishing examination sensitivity levels including instrument controls to be used.

- Scanning surface limitations, if applicable. The volumes to be examined, and the types and orientation of flaws to be detected.

- Detection and sizing techniques including angles and modes of wave propagation in the material, directions, maximum speed, minimum overlap, and extent of scanning, method of calibration, and method of establishing scanning sensitivity levels.

- Data to be recorded and method of recording.

- Method and criteria for the discrimination of indications (e.g., geometric versus flaw indication).

The document requires requalification if any essential variable is changed. If the examination procedure allows a range of any essential variables, the document requires qualification over said range.

The Code Case being developed by Section XI ensures qualification of procedures and personnel on realistic flaws. However, the number of flaws required for qualification by the document is inadequate and will not provide a statistically significant performance demonstration. Adopting the proposed appendix will only incorporate current personnel qualification requirements (i.e., IEB 83-02) into the Code.

4.8 AREAS OF RESEARCH

There are active research programs currently being pursued by the industry and the NRC. The intent here is not to describe these programs but to focus on the issues that need to be emphasized or added to the existing programs to address issues identified in this section and to provide timely resolutions.

4.8.1 Applied Research

Recent laboratory research and field experience indicates that the need for improvement is still critical for both detection and characterization (dimensioning) of cracks. Specifically, the following inspection issues need to be addressed immediately:

- Develop and validate effective and reliable manual UT inspection methods to correctly interpret UT indications.
- Develop ultrasonic techniques for dimensioning flaws in the through-wall plane.
- Develop inspection techniques for detection and dimensioning of flaws in pipe repaired by the weld overlay process.

The following issues need resolution on a less urgent basis:

- Develop reliable automated UT equipment to detect, interpret, and characterize cracks.
- Determine the effect on detection and characterization of cracks in pipe that has had the induction heat stress improvement (IHSI), or last pass heat sink (LPHS) treatments.

- Develop a practical means for implementing a "transfer method" to compensate for response differences between the calibration block and the piping material. This concept was recently eliminated from the Code requirements since no practical means existed for its implementation. However, the concept has merit, and a serious development effort is warranted.

- Develop inspection techniques for examination of austenitic butt welds through the weld metal (i.e., far-side access which is typical for a pipe-to-component weld).

- Develop reliable inspection techniques for piping with inner surface cladding, for welds in CCSS piping (for PWR) and for dissimilar metal welds.

- Establish the reliability of advanced techniques.

4.8.2 Basic Research

Additional research is needed on the interaction of ultrasound with materials and cracks. Excellent modeling work has been completed for predicting the results obtained from the Program for Inspection of Steel Components (PISC) experiments. However, these modeling efforts need to be expanded. Specifically, finite element analysis modeling should be done for cracks in austenitic (wrought and cast) material. This work should be used to predict results and complement work in:

- Developing the inspection techniques listed above.

- Predicting conservative crack characteristics that could be used in qualification test samples.

- Reliability of advanced techniques.

4.8.3 Human Factors Research

Research is needed to identify, isolate, and analyze the human factors aspects of inservice inspection, and their influence on the overall effectiveness of NDT/ISI. After these human factors aspects have been identified and quantified, they should be prioritized with respect to relative influence and relative correctability. Finally, recommendations should be developed toward mitigating the consequences of the major negative human factors aspects.

4.9 FOREIGN EXPERIENCE

4.9.1 Experience in Sweden

Because of leaks found in the 4-in.-diameter control rod drive hydraulic system of Ringhals Unit 1, UT was performed for 60 welds of 4-in.-diameter piping in April 1983 (4.5, plus private communication with

Dr. Ferenc de Kazincy of SA). Twenty-four welds were reported to contain 39 crack indications. On the basis of this finding and higher-than-normal carbon content in two heats from which the piping was made, the utility decided to replace piping containing 247 welds with 316NG material. On the cut-out pipes, only 6 of the 39 indications could be metallographically verified as being due to IGSCC. Two more cracks were discovered that were not detected by UT (one of which had initiated in the weld root). Crack lengths ranged between 10 and 70 mm. Most of the other UT indications were attributed to geometrical reflectors such as weld roots and poor penetration. Grinding on the weld ID also produced UT indications. Because of the high rate of false calls in UT, it was decided that from 1984 on operators must be trained before performing inservice UT on cut-out pipe welds which contain IGSCC.

An additional 90 welds from the cut-out pipes which had not been ultrasonically tested previously were tested by liquid penetrant and 19 crack indications were found, 15 of which were confirmed by radiography to be cracks. Further examinations are under way.

A comparison of UT response amplitude versus actual crack depth showed no systematic correlation. Length measurements, however, turned out to be rather accurate using the 50% DAC technique. The following tentative classification of the indications is being used.

1. Indications with high amplitude (300-1200% DAC) and a length of 60 mm (2.36 in.). These are judged to have a relatively high probability for causing unacceptable leakage rates.

2. Indications with a high amplitude (300-1200% DAC) and a length of 40 mm (1.6 in.). These cracks are judged to have potential for small, but acceptable leakage rates.

3. Indications with a low amplitude (200% DAC) and a length of 40 mm (1.6 in.). These cracks are judged not to be harmful during the next operating period (1000 hours).

All Type 1 UT indications are repaired, and Type 2 UT indications are repaired, depending upon the possible consequences of a leak.

The Swedish Nuclear Power Inspectorate (SKI) has initiated development projects to provide better discrimination (detection and interpretation) procedures, and to develop more reliable sizing techniques. A "consultative" group has been formed from representatives of the utilities, SKI, the Swedish Plant Inspectorate (SA), and a testing company. This group is coordinating a research program to improve UT techniques which will include:

- Evaluation of all IGSCC examinations to date.
- Qualification programs for UT operators.
- Procedures for better discrimination of cracks and geometrical indications.
- Optimization of equipment (search units with different frequency, damping, and refraction angles will be studied).

In late 1983, SKI initiated a project for investigation of the 'crack-tip-reflection-technique.'

4.9.2 Experience in Japan

In a recent Japanese paper (4.6), it was reported that five techniques were investigated during a study to measure crack depth by UT. The test specimens were 5.3-cm-thick weld joints with natural defects. The UT techniques investigated were tandem, tip echo, mode-converted surface waves, tomography, and the ALOK method (amplitude and time-of-flight locus). The results showed no significant differences between the relative capabilities of these five techniques for measuring crack depth. This study will continue until at least 1985.

The Nuclear Power Engineering Test Center (NUPEC) is conducting a program to evaluate the reliability of ISI methods (UT and ET), and to modify the inspection procedures to reduce the radiation exposure of inspectors (4.6). This major program (8 years and 12 million dollars) was initiated in 1978, and includes various full-scale mockups of portions of a variety of nuclear power plant components. The artificial defects to be introduced include EDM notches, drilled holes, fatigue cracks, weld defects, and stress-corrosion cracks. For piping, the depth of the flaws will be from 5 to 15% of the nominal wall thickness, and the flaw lengths will be 2 to 6 times the depth. Most of the flaws will be located in the weld heat-affected zone. Evaluations of flaw detectability, overall technique performance, and inspection time will be conducted for both manual and automatic UT techniques.

A comparison of the UT response from side-drilled holes and notches was conducted (4.6). These results showed that for 45° shear waves, the notches provided the greatest response; whereas, for 60° and 70° shear waves, the response was about the same for thicknesses up to about 20 mm, and the side-drilled holes provided slightly greater response when the section thickness was 30 mm. The Japanese concluded that when the UT calibration sensitivity is established using side-drilled holes, the flaw detection sensitivity will be higher for 45° shear waves and lower for 60° and 70° shear waves, compared to the sensitivity established using notches. Additional work was conducted to correlate UT response amplitudes with the area (depth x length) of defects. Although the data exhibited considerable scatter, usable correlations were obtained and it was also found that the response amplitudes were influenced more by depth than by length (with a constant area defect). The influence of applied stress on the UT response from fatigue cracks was also investigated. The echo amplitudes varied approximately 25 dB owing to tensile and compressive loading for non-oxidized crack surfaces, but this difference was only about 5 dB when the crack surfaces were oxidized (the situation they expect in an actual pipe containing high temperature coolant water).

The Tokyo Electric Power Company (TEPCO) and the Toshiba Corporation have reported on their experience at the Fukushima Unit No. 3 (4.7). Investigations were conducted to evaluate six pipe branches removed from the PLR system during the fifth outage in 1982, and the cracking was determined to be caused by intergranular stress corrosion. The response from UT indications (100% DAC or greater) were compared before and after IHSI treatments were performed (in April and May of 1981), and again in May 1982 after 7000 hours of plant operation. The UT measured length of these indications appeared to decrease during the period between the IHSI treatment and one year later. The UT response amplitudes also tended to decrease over this same time period. A comparison of UT and liquid penetrant (PT) results indicated that UT detected about 74.1% of the indications detected by PT. For large UT indications (100% DAC or more) the UT measured length was almost the same as the length measured by PT. The Japanese concluded from these results that the applied UT technique was "exact enough for indication evaluation." They also concluded that the IHSI performed in 1981 had been an effective countermeasure for IGSCC since no crack propagation or initiation was observed after an additional 7000 hours of plant operation. Their comparison of the UT and PT results showed that the UT techniques had been effective.

In a more recent informal report (4.9), the Japanese concluded that UT examination of the welds in plants completed prior to 1980 is difficult because: 1) the weld crown inhibits the UT scanning process, 2) weld suckback and ID counterbore cause UT indications which inhibit evaluation of valid UT signals, 3) the thickness and weld cross section are not accurately shown on the drawings. In view of these factors, very small search units (10 x 10 mm) are used for longitudinal wave, 45°, and 60° angle beam scanning, both parallel and perpendicular to the weld axis, and indications greater than 20% DAC are recorded. Indications that exhibit the following characteristics receive special attention: 1) indications oriented parallel to the weld axis, 2) indications in the weld heat-affected zone (HAZ), 3) high amplitude indications with lengths shorter than 100 mm, and 4) indications whose amplitude is much higher than during the previous inspection.

A frequency analysis technique is used to distinguish IGSCC from metallurgical indications at the weld fusion line, and irrelevant indications caused by the dendritic grain structure.

This Hitachi report also discussed UT sizing techniques for IGSCC (4.8). Crack length is measured using a focused, 45^{o}, shear wave search unit, and the 20 dB amplitude drop technique. The techniques being used to measure crack depth include: a) the longitudinal wave scattering technique, b) the peak echo technique (basically, crack-tip diffraction), and c) the surface-wave, mode-conversion technique. Because of overall unreliability, more than one of these two techniques are usually applied. The peak echo technique was considered effective for fatigue cracks, but was less effective for IGSCC because of the branched nature of this type of cracking.

Another Japanese report described a study to compare the ultrasonic response from artificial notches and IGSCC in pipes subjected to IHSI (4.9). The UT measurements included the conventional ASME Code techniques for measuring depth and length (i.e., recording peak amplitude response). The depth measurements were supplemented by the flaw-tip echo technique. Very little, if any, difference in UT response amplitudes was observed before and after IHSI for the EDM notches. Although similar results were obtained for the IGSC cracks, the data point spread was much greater (1.2 dB for EDM notches versus 2.7 dB for IGSCC). Similarly, few if any changes were observed for the estimated flaw length and flaw depth for either EDM notches or IGSCC, before and after IHSI treatment. Again, the data spread was greater for IGSCC than for EDM notches. These authors concluded that no significant changes in echo height or defect dimensioning had occurred as a result of the IHSI treatment based on the ultrasonic response data. Furthermore, they concluded that the flaw-tip echo technique provided an effective means for measuring flaw depth based on a correlation of UT predicted depths with destructive analysis.

4.9.3 Experience in Switzerland

The KKM plant in Muhleberg, Switzerland (a 334-MWe BWR) has been in operation since October 1972. A plant visit by an NRC staff member and an NRC consultant provided the following information.

Preservice examination of the piping system involved only dye penetrant and radiography - a UT baseline was not conducted. The initial construction process did not include grinding of the weld crowns; however, it is planned to manually grind all crowns for future UT exams. Their experience to date has shown that the shop welds contain very few UT indications, but the field welds contain many UT indications.

Throughwall leaks were visually detected in the 4-in. bypass lines during routine post-outage pressure tests (to operating pressure) before startup. The throughwall cracks in the 4-in. bypass lines were found by destructive evaluation to be 2 to 3 in. long. In 1980, all 4-in. bypass lines were removed and 304L stainless steel end caps were installed.

Several UT indications (interpreted to be cracks) have been found adjacent to welds on the pump discharge line. These cracks appear to extend 360° around the pipe, and vary from 2 to 6 mm in depth. After 3000 hours of plant operation, another UT inspection was performed and no detectable crack growth was found. They plan to operate until August 1984, at which time the pipe will be replaced with 316NG stainless steel pipe.

Currently, all UT inspections in Switzerland are performed manually. Sensitivity calibrations are based on a 2-mm side-drilled hole. Depth sizing is performed using various techniques, and the information from each technique is compared before a final sizing call is made. In addition, they do not feel that effective examinations are being performed from the component side of static cast components such as elbows and valves.

4.10 LEAK DETECTION IN LWRs: REVIEW AND RECOMMENDATIONS

4.10.1 Introduction

Early detection of leaks in nuclear reactors is necessary in order to detect deteriorating or failed components and minimize the release of radioactive materials. Consequently, the NRC requires that operational leak-detection systems of various kinds be installed. Since equipment cannot be perfectly leak-tight, allowance is made for identified leakage from valve packing, shaft seals, and other equipment. Thus, even during normal operation, there may be some accumulation of water in the sumps with an increase in the level of radioactivity. For most BWRs the plant technical specifications require that plant shutdown be initiated for leak rates of 5 gpm (unidentified) or an increase of 2 gpm in 24 h (unidentified).

4.10.2. Standard Practice

The American National Standards Institute (ANSI) has prepared a draft standard (4.10) which reviews several leak-detection methods and indicates their capabilities for detection, location, and measurement. This information is summarized in Table 4.1.

As Table 4.1 indicates, no single currently used leak-detection method combines optimal leakage detection sensitivity, leak-locating ability, and leakage measurement accuracy. Consequently, Regulatory Guide 1.45 suggests that at least three different detection methods be employed in the reactor. Sump flow and airborne particulate radioactivity monitoring are mandatory. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous activity. Although the current methods used for leak detection reflect the state of the art, other techniques may be developed and used. Regulatory Guide 1.45 also suggests that flow rates from identified and unidentified sources should be monitored separately to an accuracy of 1 gpm, and indicators and alarms for leak detection should be provided in the main control room.

Because the recommendations of Regulatory Guide 1.45 are not mandatory, the technical specifications for 74 operating plants (including PWRs) have been reviewed to determine the type of leak detection methods employed, the range of limiting condition for operation, and the surveillance requirements for the leak detection systems.

Table 4.1 Capabilities of Leakage Monitoring Methods (from Reference 4.10)

Method	Leakage Detection Sensitivity	Leakage Measurement Accuracy	Leak Location
Sump Monitoring	G[a]	G	P[c]
Condensate Flow Monitors	G	F[b]	P
Radiogas Activity Monitor	F	F	F
Radioparticulate Activity Monitor	F	F	F
Primary Coolant Inventory[d]	G	G	P
Humidity Dew Point	F	P	P
Tape Moisture Sensors	G	P	G
Temperature	F	P	F
Pressure	F	P	P
Liquid Radiation Monitor[e]	G	F	F
Visual[f]	F	P	G

[a]G (Good) - can generally be applied to meet intent of this standard if properly designed and utilized.

[b]F (Fair) - may be acceptable, marginal, or unable to meet intent of this standard depending upon application conditions and the number of measurement points or locations.

[c]P (Poor) - not normally recommended but might be used to monitor specific confined locations.

[d]For PWR during steady-state conditions.

[e]For detection of intersystem leakage; may also be used for location function in sump or drain monitoring.

[f]Provided that the leakage area is visible.

All plants use at least one of the two systems recommended by Regulatory Guide 1.45. All but eight specify sump monitoring as one of the leakage detection systems, and all but three use particulate radioactivity monitoring. Monitoring drywell air cooler condensate flow rate and atmospheric gaseous radioactivity are also frequently used. Leakage limits for most plants have also been tabulated. The allowed limits on reactor unidentified coolant leakage are shown in Figure 4.1a. The limit for all PWRs is 1 gpm and the limit for most BWRs is 5 gpm. The limits for total leakage (Figure 4.1b) are generally 10 gpm for PWRs and 25 gpm for BWRs. (Regulatory Guide 1.45 does not specify leakage limits, but does suggest that the leakage detection system should be able to detect a 1-gpm leak in 1 hour.) In some cases limits for rates of increase in leakage are stated in the plant technical specifications. On an hourly basis they are either 0.1 gpm/h (2 BWRs) or 0.5 gpm/h (4 BWRs). Additional limits for rates of increase in leakage (2 gpm/24h) were temporarily imposed on five BWRs as part of the five orders (IGSCC inspection orders confirming shutdown) of August 26, 1983.

Surveillance periods are indicated in Figure 4.2a. Leakage in most PWRs is checked every 12 hours, and in most BWRs every 4 or 24 hours. One BWR specifies that a continous monitor with control room alarm shall be operational. The intervals between system calibration and checks are indicated in Figure 4.2b. For BWRs, calibration is generally performed at 18-month intervals and functional tests every month.

In general, sump pump monitoring is used to establish the presence of leaks. Other methods appear to be less reliable or less convenient. In most reactors the surveillance periods are too long to permit detection of a 1-gpm leak in one hour as suggested by Regulatory Guide 1.45, but it appears that this sensitivity could be achieved if monitoring procedures were modified. None of the systems provide any information on leak location, and leaks must be located by visual examination after shutdown. Since cracks may close when the reactor is shut down, reducing flow rates considerably, it would be desirable to be able to locate cracks during plant operation.

The estimated sensitivity of leakage monitoring systems is occasionally addressed in the technical specifications. For example, one specification indicates that air particulate monitoring can in principle detect a 0.013-gpm leak in 20 min., that the sensitivity of gas radioactivity is 2-10 gpm, and that of condensate flow monitoring is 0.5-10 gpm. Sump pump monitoring appears capable of detecting a 1-gpm leak in 10-60 minutes (with continous monitoring).

The impact of Reactor Coolant Pressure Boundary (RCPB) leakage detection systems on safety was evaluated for 8 reactors as part of the Integrated Plant Safety Assessment-Systematic Evaluation Program (NUREG 0820-0827). In 4 of the 8 reactors a 1-gpm leak would not be detected in 1 hour nor did they have three leakage monitoring systems, as suggested by Regulatory Guide 1.45. The fracture mechanics and leak rate calculations in the SEPs are consistent with the studies reported in Section 6, which indicate that current leak detection systems and leakage limits will detect and require plant action for through-wall cracks of 4 to 10 in. in length in 12 to 28-in.-diameter piping in one day. Since these cracks are much smaller than those required to produce failure in tough reactor piping, improved leak detection systems may offer little safety benefit for this particular class of flaws when crack growth occurs by a relatively slow mechanism. However, the SEPs note that local leak detection systems may be desirable for some postulated break locations where separation and/or restraint is not practical to remove the effects of a high energy pipe break.

Although current leak detection systems are adequate to ensure leak-before-break in a great majority of cases, the possibility of large cracks resulting in small leaks must be considered. This could arise because of the relatively uniform growth of a long crack before penetration or the corrosion plugging or fouling of relatively slowly growing cracks. In such cases the time from a small leak to a significant leak or rupture could be short depending on crack geometry, pipe loading, and transient loading (sesmic or water hammer event).

The shortcomings in existing leak detection systems are not simply a matter of conjecture. The Duane Arnold safe end cracking incidents indicate that the sensitivity and reliability of current leak detection systems are clearly inadequate in some cases. The plant was shut down on the judgment of the operator when a leak rate of 3-gpm was detected; however, this rate is below the required shutdown limit for almost all BWRs. Examination of the leaking safe end showed that cracking had occurred essentially completely around the circumference. The crack was throughwall about 20% of the circumference and 50-75% throughwall in the nonleaking areas. The other seven riser safe ends were also severely cracked, but since the cracks were not throughwall no leakage resulted.

Simply tightening the current leakage limits may not be an adequate solution to these shortcomings, since it is possible that this may produce an unacceptably high number of spurious shutdowns due to the inability of current leak detection systems to identify leak sources.

One other safety-related aspect of improved leak detection systems is in the area of radiation exposure to plant personnel. Improved systems with leak location capability could reduce the exposure of personnel inside containment and present an attractive alternative to augmented ISI. Some welds are inaccessible either for inspection or replacement, and improved leak detection would provide additional margin against possible failure in these locations. Improved leak detection is consistent with the defense-in-depth philosopy of the NRC and would lead to earlier detection of system degradation.

4.10.3 Improved Leak Detection Systems

In order to improve detection of leaks through IGSCC, some utilities have installed either acoustic emission monitors or moisture-sensitive tape (MST) for better localized leak detection at specific welds. Acoustic monitors have been installed at Dresden (Commonwealth Edison) and Hatch (Georgia Power Co.). MST has been installed at Browns Ferry-2 (TVA), Peach Bottom 2 and 3 (Philadelphia Electric), and Vermont Yankee (Vermont Yankee Nuclear Power).

In general, MST or acoustic emission monitors (AEM) have been installed near welds that have crack indications. At Peach Bottom, a MST sensor in the vicinity of a leaking valve triggered an alarm. No other incidents of leak detection by MST or AE in-reactor have been noted at the time of this report. MST systems are sensitive if a path exists for leakage to reach the detector; however, the incident at Peach Bottom points out the need for quantitative information regarding leak characterization, location, and flow rate to avoid false alarms which lead to unscheduled shutdowns and unnecessary exposure to plant personnel during subsequent inspection inside containment.

No leak has been indicated by the AE system installed at Hatch and no leaks have been found during shutdown periods. Tests on a virtually identical system at Argonne National Laboratory indicate that because the acoustic background level is very low in this particular case, leaks as small as 0.002 gpm could be detected. However, the system has a limited dynamic range, saturating at approximately 0.006 gpm. Thus it could also be prone to alarms generated by relatively large leaks from sources other than cracks far from the crack location being monitored (an instance might be valve leakage).

High temperature piezoelectric accelerometers have been used at the Dresden 2 reactor on safety relief valves and on a main recirculation line (28 in.) elbow. The system detects signals from leaks in the 20 to 50-kHz range and employs a spectrum analyzer to verify if a leak is present. Signals in a specific frequency window suggest the presence of a leak. At the time of this report no alarm signals have been detected. This system is considered experimental and the main leak detection system is still the sump monitor. High temperature accelerometers have also been employed by Philadelphia Electric (PECO) since 1974 to monitor valves for leakage. The primary cause of plant shutdown has been packing gland valve leaks. Leaks as low as 0.5 gpm can be detected by the PECO system.

Several plants in Sweden and Finland have installed television monitors in containment to aid in identifying and locating leaks. Although lacking in sensitivity and completeness of coverage, such systems appear to be useful, inexpensive supplements to current leak detection systems.

Although current installations probably do not fully utilize the best available technology, the use of acoustic monitoring or MST to improve leak detection capability at specified sites is technically feasible and has been studied by a number of investigators (4.11-4.17). However, current acoustic monitoring techniques provide no source discrimination (i.e., pipe crack vs. valve leak) and no flow rate information (i.e., the system may saturate on a small leak). MST does not provide quantitative leak rate information and gives no specific location information other than the location of the tape; moreover, its usefulness with the new porous "soft fiberglass" insulation which permits escape of steam through the insulation, needs to be demonstrated.

Leak detection techniques need further improvement in the following areas: 1) improving reliability and eliminating false calls through location information and leak characterization to identify the source; 2) quantifying and monitoring leak rates; and 3) minimizing the number of installed transducers required for a "complete" system (see also reference 4.17).

A number of studies have been carried out to evaluate the feasibility of using acoustic leak detection to achieve these goals. Smith et al. (4.11) have studied acoustic leak detection in PWRs. Both field tests and laboratory studies were conducted. The results suggest that leaks as small as 0.02 gpm could be detected (near the source) in a PWR by acoustic emission techniques, and that acoustic emission monitoring of valve and pipe leakage would be useful. They suggest that valve leakage could be located by measuring the amplitude decay of the resultant acoustic signal. Researchers at Battelle-Columbus (4.12) have studied the acoustic signals generated by the flow of initially saturated or subcooled liquid through narrow slots and laboratory-grown IGSCC. Acoustic data have been acquired from several transducers for flow rates of approximately 10^{-4} to 1 gpm. Studies at KWU, Erlangen, West Germany (4.14, 4.15), have also tried to establish the sensitivity of acoustic leak detection in the presence of acoustic background noise. These studies indicate that it is possible to detect leaks in primary loops of a PWR with a leak rate of approximately 0.4 gpm by acoustic means, leak location on a specific component is possible, and relatively few transducers are needed to monitor a loop for leaks. A prototypical system will be tested on a primary loop. The system is being considered as a redundant alternative to ISI. Acoustic leak detection has also been studied at the Japanese Central Research Institute of the Electric Power Industry Energy and Environment Laboratory (CRIEPI (4.15, 4.16). They have concluded that acoustic leak detection is useful for detection of leakage from boiler tubes and that the techniques are applicable to piping in nuclear power plants.

The ongoing U.S. NRC program on leak detection at Argonne National Laboratory (ANL) is directed at the development of an acoustic leak detection (ALD) system for complete system monitoring. The program will determine whether meaningful quantitative data on leak rates and locations can be obtained from acoustic signatures of leaks from IGSCCs and fatigue cracks in low and high pressure lines, and whether these can be distinguished from other types of leaks. The program will also establish calibration procedures for acquiring acoustic data and will determine whether advanced signal processing can be employed to enhance the adequacy of ALD schemes. Based on the laboratory measurements at the Hatch and Watts Bar reactors, estimates of the sensitivity of acoustic leak detection systems can be made (4.18). Under subcontract to ANL, GARD Inc. has established a system configuration and developed a breadboard system, which will provide a basis for a prototype leak detection location, and quantification system. The breadboard system will be evaluated in FY 1984.

Complete monitoring systems are desirable to provide additional margin for undetected or unsuspected cracking while avoiding spurious shutdowns. The installation of television monitoring may be a relatively simple, worthwhile addition to current leak detection systems. Complete monitoring systems are not yet available. However, prototype ALD systems and MST systems are being developed. Testing of these prototype systems in-reactor to determine the reliability of hardware and accuracy of these techniques for leak detection should be encouraged.

4.11 CONCLUSIONS AND RECOMMENDATIONS

Previous subsections in this section have described significant problem areas relative to the uncertainty of UT inspection during both preservice and inservice examination. The following items in this subsection represent a synthesis of important conclusions and recommendations.

- Code minimum UT procedures result in totally inadequate IGSCC detection. Easily implementable modifications to these procedures have resulted in some improvement. These have been incorporated into Code Case N-335. Therefore, it is recommended that Code Case N-335 should immediately be made mandatory for all augmented inspections until better procedures are developed.

- Although IGSCC detection has improved to the point that it is considered acceptable under optimum conditions and procedures, the detection reliability as impacted by variability in operator procedure and equipment performance along with field conditions needs further study and improvement. While length sizing of cracks is acceptable, depth sizing is currently inadequate. It is recommended that advanced techniques and procedures for crack detection and depth sizing continue to be developed and incorporated into Code requirements to provide data to reduce the need for extremely conservative fracture mechanics evaluation.

- The current activities in personnel and procedure qualification and performance demonstration represent steps in the right direction, and the resultant process that is being implemented is acceptable in the interim; however, they need further improvement. Therefore, it is recommended that ongoing industry NRC activities to develop adequate criteria for qualification of the entire inspection process to achieve more reliable field inspection be completed and implemented on a high priority basis.

- For future plants or for replacement of existing piping systems, the material, design of pipe joints, and accessibility from both sides of the weld should be optimized for UT examinations; this requirement should be mandatory for all components with the exception of existing items such as pumps, valves and vessels in older plants. The uninspectable joints should be subjected to IHSI.

- Inspection techniques should be developed for detection and dimensioning of flaws in pipes repaired by the weld overlay process.

- Improved leak detection systems would permit more stringent requirements on unidentified leakage without increasing the occurrence of spurious shutdowns due to relatively benign leakage, and their development should be pursued.

- Acoustic and moisture-sensitive tape leak detection systems for local leak detection are available, and their use is recommended where inspection is difficult or impossible or operation for more than one fuel cycle is considered for long cracks with weld overlay repairs.

- An effort is also needed to be made to accumulate information on current acoustic leak detection (ALD) and moisture sensitive tape (MST) field installations. Detailed reports on the operability, maintenance, and reliability of these systems should be acquired, assembled, and distributed for review by utility, government, and research personnel interested in improving reactor leak detection technology.

- Sump pump monitoring systems have sufficient sensitivity to detect leak rates as small as 1 gpm in one hour. Improved leak detection systems would provide additional margin against leak-before-break without increasing the number of false alarms by providing information about leak location and leak source discrimination. It is, therefore, recommended that improved leak detection systems under development be completed and field tested.

- Since the major purpose of the preservice inspection (PSI) is to provide a reliable NDE baseline for comparison with subsequent ISI, it is recommended that the latest edition of Section XI acceptable under CFR 50.55(a) be used for the PSI.

- Expand modeling work to predict UT crack response for guiding development of UT techniques and guiding qualification test sample selection.

- Since the UT examiner is one of the more erratic inspection variables, it is recommended that human factors research be performed to lead to a reduction in the human variability.

In the event that any of the preceding recommendations of this section are implemented there may be a need to modify the following documents when relevant to the specific issue:

NUREG-0800, Sections 5.2.4 and 6.6; Draft NUREG-0313, Revision 2; Generic Safety Issue 14, A-42 and A-14.* ASME Section XI; Regulatory Guide 1.45; and the development of new regulatory guides on qualification of UT personnel, equipment and procedures, and on pipe inspection.

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*This has been dropped as a Generic Safety Issue because it is not a safety issue per se, but it impacts specific problems which the Office of Nuclear Regulatory Research is addressing.

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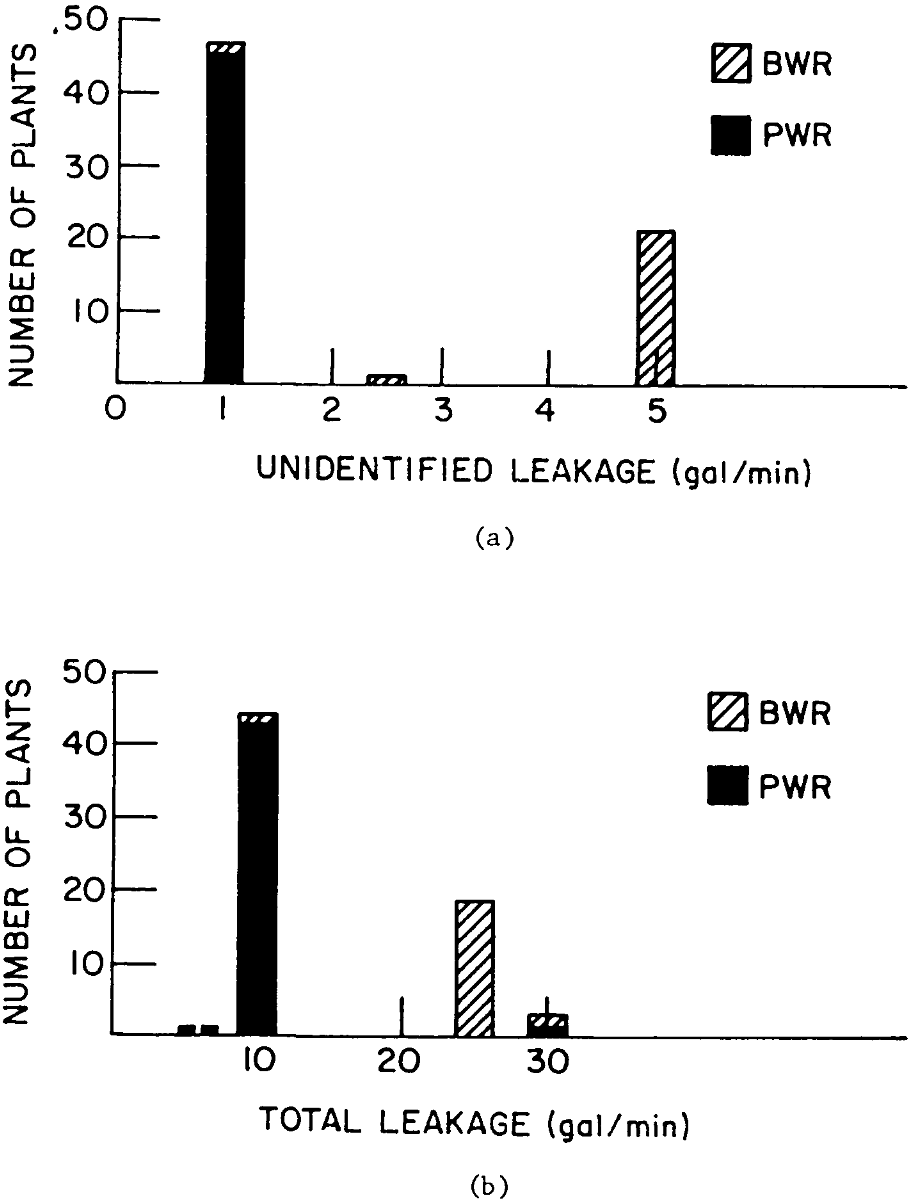
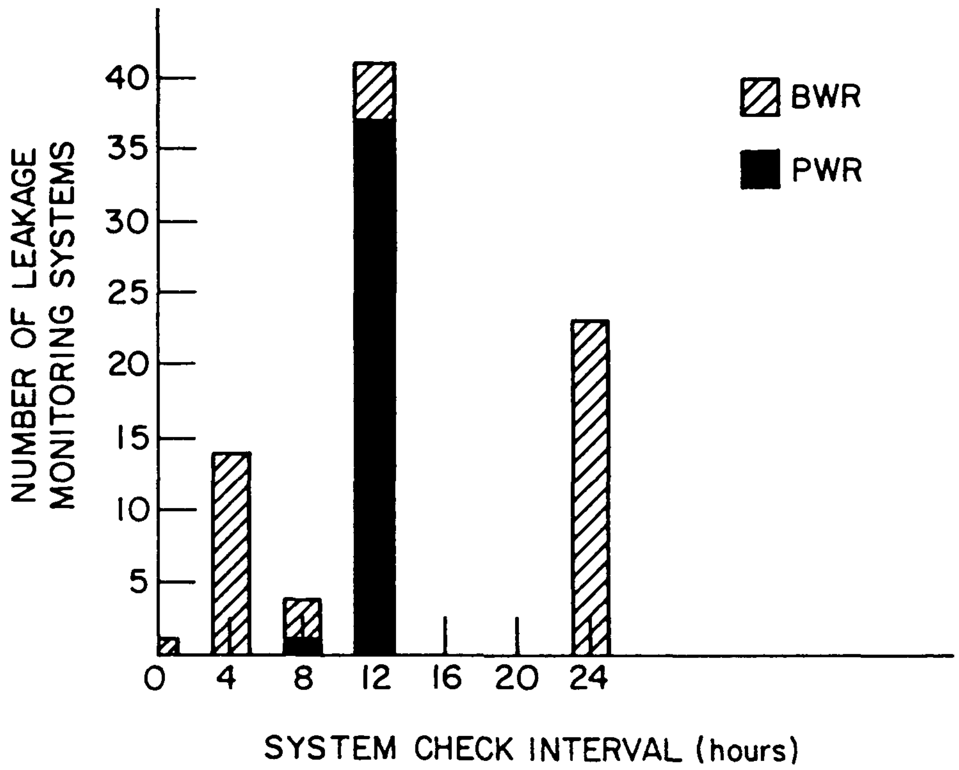
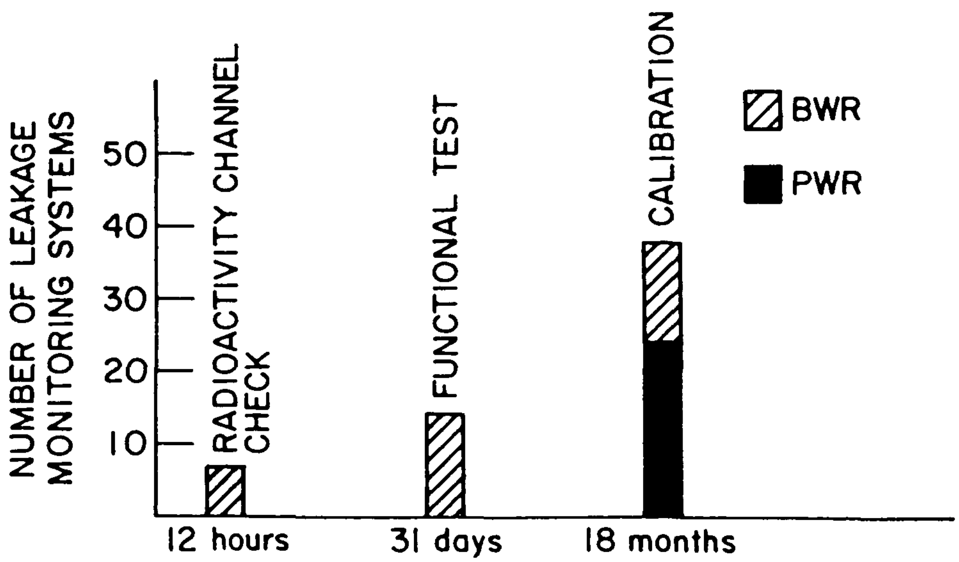


Figure 4.1 Allowed Limits on Reactor Coolant System Leakage from Technical Specifications for 74 Plants for a) Unidentified Leakage and b) for Total Leakage



(a)



(b)

Figure 4.2 Histograms Based on the Technical Specifications for 74 Plants and the Number of Leakage Monitor Systems Versus a) System Check Interval (hours) and b) System Calibration or Functional-Test Interval

5.0 INSPECTION OF PIPING FOR IGSCC

5.1 INTRODUCTION

The extent and frequency of inspection of susceptible piping welds recommended in this section are based primarily on the reinspection program specified in SECY-83-267c (5.1) for those plants inspected under IE Bulletins 82-03 and 83-02 (5.2, 5.3), and the inspection programs recommended in the draft NUREG-0313, Rev. 2 (5.4). The reinspection program specified in SECY-83-267c was reviewed and approved by this task group and is included in Sub-section 5.4.2 for completeness because these plants will be reinspected within the next year or so. This task group has also reviewed the overall inspection program presented in the draft NUREG-0313, Rev. 2, and has developed the recommended inspection program in this section.

One positive result of the extensive inspections performed on larger-diameter ($\geq$ 4 in.) BWR piping is that no other significant mode of degradation has been noted. This means that inspections can focus on those approaches best suited for detecting and evaluating IGSCC. A less favorable finding is that special methods and specific operator training are required to detect IGSCC reliably. Further, differentiating UT signals arising from geometric configurations from signals caused by cracks is perhaps even more difficult than detection of indications.

It is not the intent of this section to provide specific guidance to operators regarding details of equipment and procedures. This function is best handled by Code activities in which industry and regulatory participants reach a consensus. It is not a simple problem; finding and recognizing IGSCC by UT is still as much an art as it is a science. UT examiners must also characterize quantitatively the cracks they find. IWB 3600 rules for flaw evaluation (ASME Code, Section XI) are based on the assumption that the depth and length of flaws are known reliably. Although the length (circumferential extent) of cracks is believed to have been determined with sufficient accuracy and reliability by inspectors capable of detecting cracks, accurate measurement of the depth (throughwall dimension) cannot be relied upon. While some UT examiners may do an adequate job, most will require special training to ensure sufficient accuracy for use as the basis for Code flaw evaluations.

Nevertheless, the experience gained during the inspections performed under IE Bulletins 82-03 and 83-02, UT sizing round robin tests, and particularly the insight into the problems gained through the performance capability demonstrations specified by the bulletins, can be used to formulate requirements that will significantly upgrade the reliability of future inspections.

The intent of the recommendations covered by this section is to ensure that (1) an adequate sample of susceptible pipe welds will be inspected on a sufficiently frequent basis to detect IGSCC before it can jeopardize the integrity of the piping system and (2) the UT operators inspecting BWR piping for IGSCC can reliably detect and recognize IGSCC in the welds they inspect.

5.2 INSPECTION RECOMMENDATIONS

All BWR Types 304 and 316 piping systems operating at temperatures over 200°F (93°C) have shown susceptibility to IGSCC; therefore, all systems are considered to be equally prone to cracking, and the recommendations for augmented inspections beyond those required by ASME Code, Section XI, will apply to all Type 304 and 316 austenitic piping operating over 200°F (93°C). Differences in the extent and frequency of examinations will depend on the resistance to IGSCC of the materials and the effectiveness of any processing used to prevent cracking.

5.2.1 Categorization of Welds

All welds in BWR systems will be categorized according to how likely they will be to crack. Three categories will be used:

Category A - Welds very unlikely to have IGSCC, because the piping is made of resistant materials, or welds made with, or subjected to, a countermeasure A mitigating process.

Category B - Welds with some degree of improved resistance to IGSCC, because the piping is not made of resistant material, and welds are made with or subjected to a countermeasure B mitigating process.

Category C - Welds likely to be subject to IGSCC because they are neither made of resistant material nor subjected to a mitigating process.

5.2.2 Bases for Weld Categories

The bases for categorizing welds are directly related to the materials and the welding or mitigating processes used. These are summarized below:

Resistant Materials

(1) 304L, 316L, 316K, 304NG, 316NG, 347NG, 308L

(2) low-strength carbon steels

(3) approved nickel-based materials

(4) cast low-carbon/high-ferrite austenitic stainless steels

(5) welds solution heat-treated after fabrication and welding

(6) other, as approved by NRC

Countermeasure A Processes - Any combinations of two mitigating processes which are intended to reduce or minimize any two of the three causes (i.e., sensitization, stress and environment) contributing to IGSCC. These are summarized below:

(1) IHSI on new pipe or pipe with no reported indications plus hydrogen water chemistry*;

(2) HSW on new welds, including new welds used to install short repair sections ("pup pieces") plus hydrogen water chemistry;

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*Hydrogen water chemistry is discussed in Section 6.

(3) LPHSW on new pipe welds including new welds used to install short repair sections plus hydrogen water chemistry.

(4) Other, as approved by NRC.

<u>Countermeasure B Processes</u> - Any of the following mitigating processes.

(1) IHSI on new pipe or pipe with no reported indications.

(2) HSW on new welds including new welds used to install short repair sections ("pup pieces").

(3) LPHSW on new pipe welds including new welds used to install short repair sections.

(4) Hydrogen water chemistry.

(5) Other, as approved by NRC.

5.2.3 Sample Selection

Results of inspections conducted to date under IEB 82-03 and 83-02 indicate that all Types 304 and 316 stainless steel piping welds in systems operating over 200°F (93°C) are susceptible to IGSCC. In addition, field data show that the cracking experience does not correlate well with the Stress Rule

Index (SRI) or the carbon content. Therefore, the primary basis for sample selection should be field experience; where other factors such as weld prep, excessive grinding, extensive repairs, or high-stress locations are known to exist, they should also be considered in the sample selection.

The results of inspections conducted under IEB 83-02 show that the specified sampling scheme (i.e., an initial sample size of about 20% in each pipe size and the logic for expanding the inspection sample size when cracks are found) has been found to be reasonably effective. For example, because of expansion of sample size under the bulletin, the 83-02 plants averaged an inspection of 62% of welds.

By specifying a limited initial sample, the inspection resources in terms of occupational exposures can be used more effectively. As more experience is gained in detecting IGSCC, the staff anticipates that this sampling scheme would be even more effective than in IEB 83-02. Allocation of examiner resources must always be made in determining the initial sample size so that the inspection resources are available, not only for the detection and the necessary sample expansion but also for the depth sizing and post-replacement baseline inspections.

5.2.4 Inaccessible Welds

For those categories B and C welds that are not inspectable because of physical access or poor inspectibility, an appropriate local monitoring system such as a leak detection system, should be employed to ensure the continuous integrity of the weld. These local monitoring systems should be approved by NRC before field applications.

5.2.5 Summary of Inspection Recommendations

Inspection requirements are dependent on the material and processes used for each weld. The piping categories and inspection required thereon are presented in Table 5.1.

5.3 INSPECTION PERSONNEL

The recent problems regarding the capability to detect and size IGSCC in BWR piping, in accordance with Section XI procedures and requirements, can be primarily attributed to either one or both factors of UT technology; namely, the ineffectiveness of UT procedures and of examination personnel.

This was clearly demonstrated by the inspection results of recirculation system piping at Nine Mile Point Unit 1, and more recently by the results of IE Bulletins 82-03 and 83-02 performance capability demonstrations, PNL round robin tests, and EPRI sizing studies. Both UT procedures and examiners are responsible for the recent poor showing of UT reliability. However, the fact that there was a large disparity in the demonstrated performance capability among inspection teams or examiners highlights the need to address the issue of the qualification and certification of NDE personnel in general, and UT examiners in particular.

5.3.1 Personnel Qualifications

Many very experienced and competent UT examiners have had little or no experience with interpreting IGSCC signals. They may not be adequately trained in this very specialized area. Still, they are "qualified" under

Table 5.1 Condition and Inspection Required for Weld Categories

Weld Category	Condition	Inspection Required
A	Resistant material or countermeasure A applied	25% of the welds of each pipe size in 10 years. At least one-third of these should be inspected every three and one-third years or the nearest refueling outage.
B	Nonresistant material with countermeasure B applied	50% of the welds of each pipe size in 10 years. At least one-third of these should be inspected every three and one-third years or the nearest refueling outage.
C	Neither of the above and all welds with detectable cracking regardless of the use of mitigating processes	100% in 6 years. At least one-half of these should be inspected every three and one-third years or the nearest refueling outage. For plants older than 6 years, all uninspected Category C welds shall be inspected at the next outage after compliance with IE Bulletin 82-03 or 83-02.*

*Weld inspections not performed in accordance with IE Bulletin 82-03 or 83-02 are considered inadequate; therefore, no credit is given for such inspections.

governing Code requirements to perform examinations of BWR piping. They may be very reluctant to take tests to prove their competence (which they feel should be taken for granted), and may also find it somewhat demeaning to be required to take special training. Nevertheless, no UT examiner should be permitted to perform inspections of BWR piping without proving his competence, even if it requires him to take special training to gain the specific skills and knowledge required to perform these inspections. The performance demonstrations specified in the bulletins were major steps in the direction required. They set a precedent, but more importantly, they made clearly evident the need for such actions. The utilities quickly understood the problem and were cooperative and supportive. They did not want to pay for inspections that were ineffective and certainly would not want to shut down in midcycle to repair a leak in a weld recently pronounced to be sound.

Inspections will continue to emphasize crack detection and discrimination, in addition to the anticipated improvement in the sizing capability. All level 1, 2, or 3 UT examiners performing operations (general scanning observations and discrete signal interpretation and sizing) should demonstrate their field performance capability in a manner acceptable to the NRC. In addition, all examiners performing evaluations must be able to view on CRT display for the entire time that the transducer is in contact with the pipe for scanning, either in real time, remotely, or on tapes.

5.3.2 Performance Demonstration Tests

It is clear that the performance capability demonstration programs similar to those conducted under IE Bulletins 82-03 and 83-02 must continue. It is also clear that the NRC now can strengthen the program in several ways. Experience has also shown that the NRC must upgrade the program if results of inspections are to be relied upon. In addition, a similar program must be developed for sizing cracks once they are detected.

Details of an upgraded program should be worked out with industry, as the original program was with representatives from the Office of Nuclear Reactor Regulation (NRR), the Office of Inspection and Enforcement (IE), the Office of Nuclear Reactor Research (RES), and all NRC Regions involved. The IE and Region-based inspectors who must ensure that inspections are properly performed must be directly involved in working out requirements and monitoring actual performance demonstrations.

Although all aspects of an upgraded program cannot be covered in this section, there are several important aspects to be highlighted.

(1) Additional suitable samples of service-induced cracked and uncracked pipe should be obtained and properly characterized as to crack depth and length.

(2) The grading of performances should be tightened up, particularly in the allowance for detection and discrimination (false calls) and sizing uncertainties (undersizes and oversizes). A false call is calling an indication a crack if it is not. A miscall is falsely identifying a real crack as a geometric discontinuity.

Performance should be judged on the basis of the importance of the call. For example, no crack should be missed (not reported) if it is large enough to require evaluation or repair in accordance with the requirements given later on in this section. Further, severe penalties should be assessed if a nonexistent crack is reported as large enough to require evaluation or repair. Flaws with the throughwall thickness less than that given in Table IWB-3514-2 of Section XI Code are considered to be acceptable without evaluation or repair.

5.3.3 Availability of Inspection Personnel

Since the issuance of IE Bulletin 82-03, the EPRI NDE Center has been conducting an IGSCC detection and discrimination training course using cracked samples. Consequently, the number of qualified IGSCC examiners, i.e., those people who have successfully completed the performance demonstration test required by IE Bulletin 83-02, has been steadily increasing. However, there is also a sign that the NDE industry is losing a substantial number of qualified UT examiners because of a labor dispute. In addition, after a few trial courses on sizing, the EPRI NDE Center is planning to initiate a formal training and qualifying course in the near future for crack sizing.

Because of the continued efforts of the EPRI NDE Center, it is expected that sufficient qualified inspection personnel will be available for the reinspection program required by SECY-83-267C. However, in the long term, the allocation of qualified UT examiners for detection, depth sizing, and post-replacement baseline inspections should be considered in all future inspection programs.

5.4 PLANT INSPECTIONS

All BWR pipe weld examinations should be performed in accordance with the latest version of Code Case N-335 and with the specific equipment and procedures used, and personnel passed in the performance demonstration tests.

BWR pipe weld examinations should be monitored by Regional Inspectors, who must be satisfied (1) that the inspection is performed satisfactorily and (2) that equipment and procedures used and personnel have passed the performance demonstration tests.

In the event that a Regional Inspector is not satisfied with the adequacy of the inspection, the inspector may require that an additional independent qualified inspection agency perform a total or partial reinspection. This may include check inspections by NRC personnel.

5.4.1 All Plants

The extent and frequency of examinations for all operating BWR plants containing Types 304 and 316 austenitic piping operating over 200°F (93°C) should follow the inspection recommendations specified in Section 5.2 of this report. The future inspection program after reinspection for those plants that are required to be reinspected in accordance with Section 5.4.2 of this report should also follow the inspection recommendations specified in Section 5.2.

5.4.2 Plants Inspected Under IEB-82-03 and 83-02

The reinspection program discussed in this subsection is taken from SECY-83-267c and is included here for completeness because the affected plants will be reinspected within the next year or so.

The scope for the inspections which follow the Bulletins 82-03 and 83-02 inspections should include the following stainless steel welds, susceptible to IGSCC, in piping equal to or greater than 4 inches in diameter in systems operating over 200°F (93°C) that are part of or connected to the reactor coolant pressure boundary out to the second isolation valve:

(1) Inspection of 20% (but no fewer than 4 welds) of each pipe size of IGSCC sensitive welds not inspected previously and reinspection of 20% (but no fewer than 2 welds) of each pipe size inspected previously and found not to be cracked. This sample should be selected primarily from weld locations shown by experience to have the highest propensity for cracking.

(2) All unrepaired cracked welds.

(3) Inspection of all weld overlays on welds where circumferential cracks longer than 10% of circumference were measured. Disposition of any findings will be reviewed on a case-by-case basis. Criteria for operation beyond one cycle with overlaid joints are under development.

(4) Inspection of any IHSI-treated weld which has not received post-treatment UT acceptance testing.

(5) In the event new cracks or significant growth of old cracks* are found, the inspection scope should be expanded in accordance with IEB 83-02.

The inspection requirements for these plants after the reinspection specified above should follow the generic inspection recommendations discussed in Section 5.2.

5.5 FOREIGN EXPERIENCE

The inspection program for austenitic piping for IGSCC in foreign countries is generally similar to that of ASME Code Section XI. The only major difference, at least in Japan, is that once IGSCC is detected in a given size piping system, all welds in that piping system and those welds in other size piping systems under the same environment are also inspected.

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*Significant growth of the old crack is defined as growth to a new crack size that cannot be accepted without repair for the remaining period of the current or a new cycle of operation, in accordance with the criteria in SECY-83-267C.

This extensive inspection can be accomplished without encountering much difficulty in the availability of inspection personnel because there is no widespread IGSCC problem in most of the foreign countries. In isolated cases such as in Japan, the utility was encouraged to replace the affected piping either immediately or after one cycle of operation with more corrosion resistant piping material in order to avoid unnecessary inspections.

5.6 CONCLUSIONS

- Augmented inspections beyond those required by ASME Code Section XI should apply to all Types 304 and 316 austenitic piping systems operating over 200°F (93°C) unless they have been treated with effective countermeasures.

- The degree of augmented inspection is dependent on the material and processes used for each weld. The most frequent inspections are required for welds fabricated from nonresistant material.

- Field data show that the cracking experience does not correlate well with the stress rule index and the carbon content. Therefore, the primary bases for sample selection should be field experience coupled with other factors such as weld prep, excessive grinding, extensive repairs, or high-stress locations.

.7 RECOMMENDATIONS

- All Type 304 and 316 austenitic piping systems operating over 200 $^{\circ}$F (93°C) should receive augmented inservice inspection unless they have been treated with effective countermeasures.

- The extent and frequency of examinations should depend on the resistance of materials to IGSCC and the effectiveness of any processes used to prevent cracking.

- The primary basis for sample welds selected for examination should be field experience, not the stress rule index and carbon content. Other factors such as weld prep, excessive grinding, extensive repairs, or high-stress locations should also be considered in the sample selection.

- The performance capability demonstration program used to demonstrate the effectiveness of UT equipment, procedure, and examiner combination on the service-induced cracked samples should be continued and strengthened. The effectiveness and reliability of detecting IGSCC with all future UT procedure/examiner combinations should be demonstrated on the cracked samples before field application.

- All UT examiners should attend the industry-sponsored UT detection and sizing training courses and should continue to be evaluated on the cracked samples under the witness of a third party personnel before participating in the field inspection.

- All BWR pipe weld examinations should be performed in accordance with the latest version of Code Case N-335 and with the specific equipment and procedures used, and personnel passed in the performance demonstration tests.

- An appropriate local monitoring system such as local leak detection system should be used to monitor the continuous integrity of the uninspectable welds in Categories B and C. These local monitoring systems should be approved by NRC before field application.

In the event that any of the preceding recommendations of this section are implemented there may be a need to modify the following documents when relevant to the specific issue:

A-42; NUREG-0313, Rev. 1; Generic Issue No. 34; 10CFR50, App. A 30 and 32; Regulatory Guides 1.58 and 1.147; SRP 5.2.4 and 5.2.5; 10CFR50.55a(g).

5.8 REFERENCES

1. NRC. November 7, 1983. Staff Requirements for Reinspection of BWR Piping and Repair of Cracked Piping. SECY-83-267C.

2. Office of Inspection and Enforcement (IE), NRC. October 1982. Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants. IE Bulletin No. 82-03, Rev. 1.

3. Office of Inspection and Enforcement (IE), NRC. March 4, 1983 Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants. IE Bulletin No. 83-02

4. W. S. Hazelton et al. 1984. "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." Draft Report, Rev. 2, NUREG-0313. Available from the Materials Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

6.0 DECISIONS AND CRITERIA FOR REPLACEMENT, REPAIR OR CONTINUED OPERATION WITHOUT REPAIR

6.1 INTRODUCTION

This section is a relatively complex interaction of the several factors pertaining to the acceptance or rejection of cracked weldments, short-term mitigative measures, and longer-term permanent fixes. The opening section covers fracture mechanics as it relates to the rates of growth of IGSCC, a discussion of leak-before-break, and an evaluation of the various fracture mechanics models that have been applied to cracked austenitic piping.

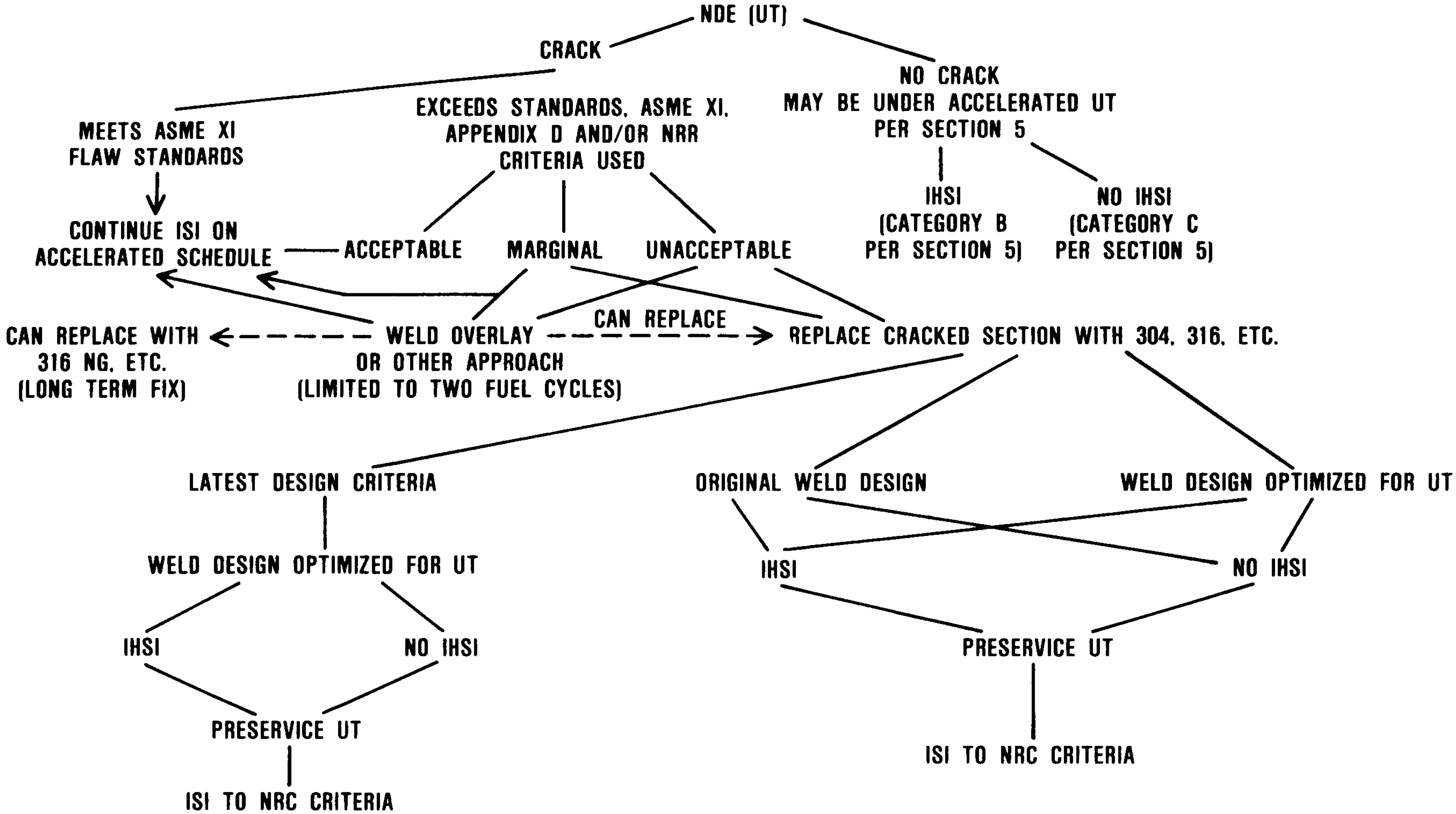
The second major subsection covers short-term measures, such as weld overlay, not considered to be a permanent fix.

The final subsection represents permanent fixes or longer-term fixes. A mitigative operation such as induction heating stress improvement falls into both short-term and long-term fixes. Controlled water chemistry is considered a long-term fix complementing mitigation; e.g., IHSI replacement of piping with a material such as 316NG is considered a long-term fix. Even so, the Task Group on Pipe Cracking considers that hydrogen water chemistry, reduction in residual stresses, and replacement with a material such as 316NG represent a desirable "belt-and-suspenders" approach.

An event tree, shown below provides an overview of the options available in the event that IGSCC is detected in the piping of a BWR. As noted in the heading, only short-term solutions are addressed in this event tree.

SHORT-TERM SOLUTIONS FOR REPLACEMENT, REPAIR OR CONTINUED OPERATION WITHOUT REPAIR

EVENT TREE*



*The following assumptions apply to the event tree:
- Repairs and replacements per original version of ASME Construction Code or, alternatively, updated to latest edition.
- Section 5 refers to this report.

The intent is to examine the options available and suggest criteria for the acceptance, conditional acceptance, or rejection of the options based on a safety assessment of each option.

The following assessments in essence consider each portion of the event tree and the regulatory significance of the various decisions.

- Detection of very small cracks at the threshold of sizing reliability would require no repair; however, the presence of these small cracks below ASME XI standard sizes, would require UT at shorter intervals until decisions could be made as to additional action. If no change in size is observed over two or three inspection intervals, the option exists to revert to the standard ISI schedule.

- If predicted crack sizes at the end of the next inspection interval are marginally acceptable, the probable action would be reassessment of the margins with weld overlay, and the application of a weld overlay. There are various potential problems with weld overlay that are addressed more fully later in this section.

- If predicted crack sizes at the end of the next inspection interval are unacceptable, various options exist, such as weld overlay, mechanical clamps, and partial replacement of the piping.

- Two options exist with regard to criteria applied on replacement, per the ASME XI Code and previous NRC policy: 1) Application of the construction version of ASME III or other Code to the replacement. This presumes a similar design assumption on loads, layout, etc. 2) Use of the latest version of ASME XI/III to update design and fabrication requirements.

- Reduction of residual stress from use of IHSI, HSW, or LPHSW should be promoted because such measures should substantially extend the time to initiation of IGSCC. Possibly, it would not initiate for the remainder of plant life.

- Material selection, if truly short term, is a second-order effect; however, available data indicate the lower levels of carbon in the L grades of 304 and 316 are highly resistant to IGSCC. A penalty exists in terms of lower stress allowables so the designer needs to consider the advantages and disadvantages of replacing with L grades and thicker piping.

- As noted in the event tree, replaced weldments require a new baseline UT examination.

- While design updating to the latest versions of the Code should not be a requirement, every effort should be made to promote its use. This should include relaxation of damping criteria on a case-by-case basis to permit a more flexible and more inspectable system.

- All steps related to the weld prep, welding, and weld finishing, and HSW or IHSI should be validated through certification and through use of an appropriate Q.A. manual. As noted previously, the weld should be optimized for future ISI.

- Future preservice and inservice examinations should utilize the advanced techniques developed under ASME XI or NRC, including certification of the NDE examiner.

- If the original stainless steel considered to be susceptible to IGSCC has not cracked but most systems have several years of operation, the leak detection systems should be reviewed with regard to both sensitivity and reliability on the assumption that delayed IGSCC may occur. If the leak detection systems are deemed inadequate, they should be improved.

6.2 FLAW EVALUATION AND LEAK-BEFORE-BREAK

The occurrence of intergranular stress-corrosion cracking (IGSCC) in operating BWRs raises two important issues related to fracture mechanics analysis. These are flaw evaluation and leak-before-break. The first of these concerns the disposition of pipes containing cracks detected during service. When a crack is detected in service an evaluation must be performed to determine if repair or replacement of the pipe is necessary or if the crack is sufficiently benign to return the cracked pipe to operation for some specified operating time. Repair of cracked pipes, generally by weld overlay, is costly in terms of both personnel exposure and extended plant outage times. Also, the weld overlay repair may induce undesirable stresses at other locations in the piping system. Thus prudent decisions must be made regarding the necessity for repairs.

The leak-before-break failure mode in piping refers to the concept that a crack propagating through the wall of a pipe, by mechanisms such as SCC or by ductile crack extension due to applied loads, will result in a stable throughwall crack that can be reliably detected by leakage. Acknowledging that some cracking which occurs may not be detected, it is desirable to demonstrate that, unchecked, IGSCC will progress in a fashion that leads to a leak-before-break mode of failure. This section presents a critical review of the currently used methods of flaw evaluation and the application of the leak-before-break concept.

6.2.1 Flaw Evaluation Criteria

The criteria currently used by the Nuclear Regulatory Commission's Office of Nuclear Reactor Regulation (NRR) for performing crack evaluations is summarized in SECY-83-267c (6.1) and presented in greater detail in draft NUREG Report 0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping-Revision 2" (6.2). The evaluation of cracks found in service requires a reasonably accurate knowledge of: 1) the ability of ultrasonic testing (UT) to size the length and depth of stress-corrosion cracks accurately, 2) the applied and residual stress states, 3) the relation of stress-corrosion crack growth rate to stress and environment, and

4) the ultimate load-carrying capacity of the cracked pipe. Section 4 of this report is devoted to the discussion of UT techniques, including crack sizing capabilities. The following two sections discuss the methods for predicting crack growth in piping weldments and determining the ultimate load carrying capacity of flawed pipes and the associated acceptable crack dimensions.

6.2.1.1 Prediction of Corrosion Crack Growth in Piping Weldments

Analyses of stress-corrosion crack growth in weldments are usually based on linear elastic fracture mechanics (LEFM) and assume that the crack growth is controlled by the stress intensity factor, K. Stress-corrosion crack growth rates have been measured in laboratory as a function of stress intensity. The stress intensity factors associated with cracks in piping can be calculated, if the crack size and shape and the stress configuration including residual and applied stresses are known. These calculated stress intensities are then combined with the measured crack growth rates to predict the growth of cracks in reactor piping systems. Direct quantitative confirmation of this approach is not available. It has been argued that it is inapplicable in the plastically deformed region near a weld (6.3). However, considering the variability in crack growth rates for different heats of material and enviromental conditions and the variability in residual stresses, the predictions of the model seem reasonably consistent with field experience and laboratory measurements on large-diameter pipes (6.9).

Since for a given weldment containing a crack the applied stresses are presumed known from the stress report, the degree of conservatism in the predicted crack growth depends on the choice of crack growth rate law, on the choice of throughwall residual stress distribution, and to a lesser degree on the assumptions used in computing the stress intensity. In the approach used currently by NRR, conservative assumptions are made for both the crack growth rate law and the throughwall residual stress distributions, and the stress intensity factor is computed assuming a complete 360° circumferential crack. Although it is conceivable that a particular weldment could have both a material with an unusually high crack growth rate and an unfavorable residual stress state, the probability of this is very low, and the overall crack growth in weldments predicted using the NRR assumptions appears to bound that observed in-reactor.

Stress-Corrosion Crack Growth Rates

In large-diameter piping the fatigue crack growth associated with design loading histories is very small, and in most cases crack growth will be due primarily to IGSCC. Cracks in smaller-diameter pipes where transients contribute more significantly to crack growth are generally repaired. The available data suggest that the contribution of the conventional design operating transients to crack growth is negligibly small (because they comprise such a relatively small fraction of the life) and that most of the crack growth occurs under the nominal steady-state operating conditions. However, it should be noted that even under "steady" load, the stresses in the piping may not be truly constant because of pressure and temperature fluctuations and mechanical vibrations. Although in most cases the associated stresses are very small and are negligible from a fatigue standpoint, small variations in stress can have significant effects on stress-corrosion crack growth (6.5). The available data for constant and "nearly constant" loads are summarized in Figure 6.1 along with the correlation currently used by NRR in the analysis of flaw growth in reactor piping.

Much of these data has been obtained on rather heavily sensitized materials in environments with 8 ppm dissolved oxygen. The sensitization level is higher than would be considered typical in as-welded material, and the test environments contain higher dissolved oxygen levels than the 0.2 ppm typical of operating BWRs. However, most of the tests have also been carried out in high purity water with impurity anion levels (sulfate, chloride, carbonate, etc.) substantially below those possible in BWRs operating under Regulatory Guide 1.56 limits on BWR water chemistry. Recent data from slow strain rate tests (6.6, 6.7) suggest that although there are significant differences in susceptibility to IGSCC in high purity environments with 0.2 ppm and 8 ppm dissolved oxygen, additions of very low levels of impurities substantially reduce these differences. Similarly the addition of impurities seems to diminish the differences in susceptibility associated with different degrees of sensitization (6.6). Even the sensitization levels may not be too conservative, since for weldments with relatively low levels of sensitization after welding, low temperature sensitization may occur during service. The uncertainties associated with degree of sensitization and water chemistries appear to be no larger than the heat-to-heat variability (e.g., the two heats of material represented by the solid triangles in Figure 6.1, which have identical heat treatments, measured sensitization values typical of weldments,

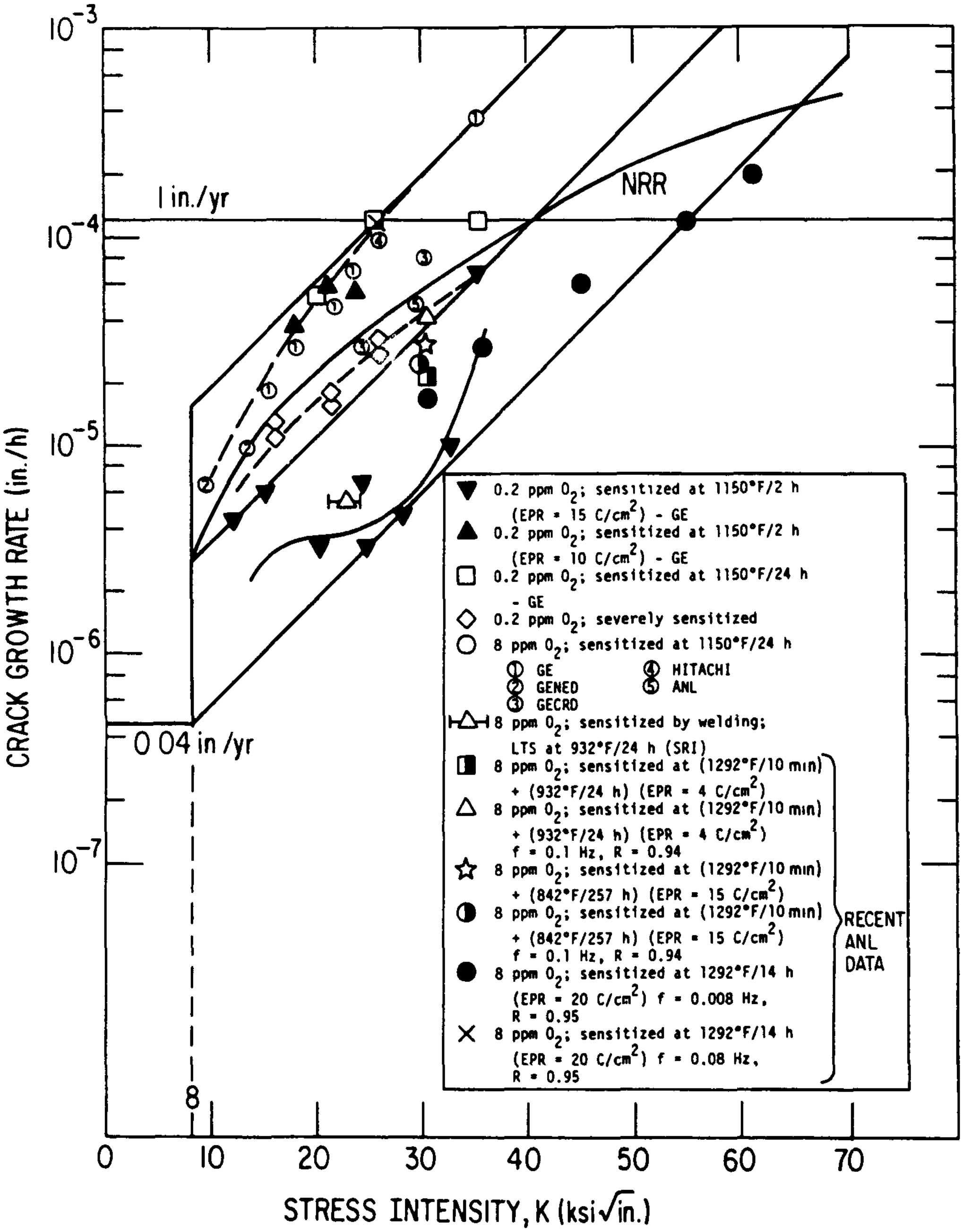


Figure 6.1 Stress-corrosion crack growth rates

and were tested in the same laboratory, give crack growth rates very close to the upper and lower bounds on all the data). Thus on the basis of the available information, the data presented in Figure 6.1 are not considered to be an unduly conservative representation of the crack growth rates expected in a reactor environment.

Current evaluations by NRR utilize the power law curve identified in Figure 6.1 which appears to be conservative for most heats of material under typical reactor operating conditions. The roll-off in crack growth rate with increasing stress intensity predicted by the power law representation does not appear consistent with the data in all cases, but this is of little consequence in the analysis of throughwall crack growth, since the stress intensity factors in this case are not large enough to require extrapolation beyond the available data.

SCC growth rates in many material/environment systems show a plateau or limiting growth rate. In mechanistic terms (6.3) a plateau could arise because there is a limiting rate for environmental attack at the crack tip. In phenomenological terms crack branching may occur which limits the stress intensity at the crack tip. In the stainless steel/BWR coolant system, the mechanistic estimates of plateau rates are very high. Crack branching is observed in both the field and the laboratory tests; however, the conditions that produce branching are not well understood. Since the mechanical loading in constant extension rate tests (CERT) is much more severe than could be encountered in any actual piping system, use of the crack growth rates observed in these tests should provide a very conservative estimate of the highest crack growth rates that could occur due to SCC in piping systems. To compute the growth of throughwall cracks for an analysis of leak-before-break the very conservative representation of the crack growth rate dependence on stress intensity by a straight line in the semi-log plot of Figure 6.1, which uses CERT data to estimate a plateau rate, is used.

Residual Stresses in BWR Pipe Weldments

Residual stresses in the weldment arise because of the thermal and mechanical loading produced during the welding process. The thermal expansion and contraction associated with the temporal and spatial variations in temperature give rise to plastic strains, and these incompatible strains produce residual stresses upon final cooling. A substantial effort has been made to experimentally measure the residual stresses in BWR pipe weldments. However, there is a great deal of variability in residual stress distributions from weldment to weldment even for large-diameter pipes. The available throughwall data for axial residual stresses in large-diameter pipes are summarized in Figures 6.2 and 6.3 (6.10). As can be seen, there is a wide range of data, presumably reflecting the variability in the welding techniques used to prepare these test weldments and the variability to be expected in the field. The distribution of residual stress used by NRR for evaluation of flaw growth is also shown in these figures. It is seen to be a conservative representation of the available data for large-diameter weldments.

Much of the information available is based on finite element calculations, since experimental measurements of residual stresses in every pipe size and weld condition of interest are not available. The experimental data serve as a "benchmark" for the finite element calculations, and the validity of the

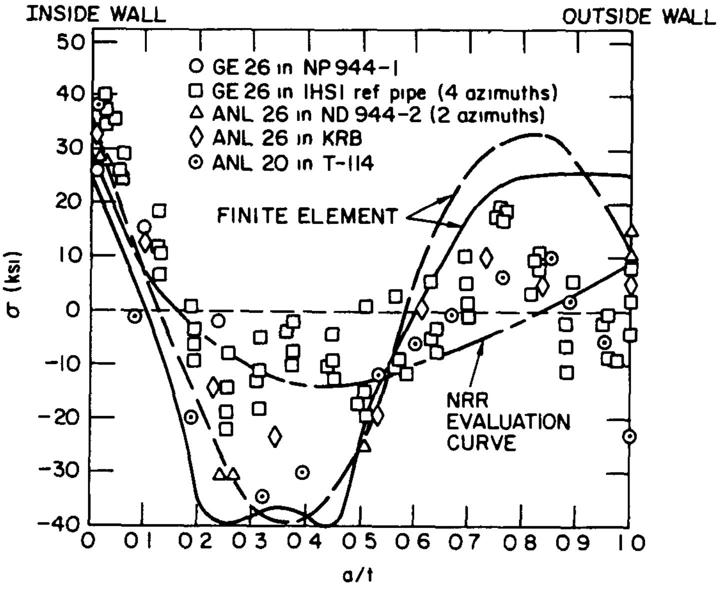


Figure 6.2 Throughwall distribution of axial residual stress in large-diameter weldments.

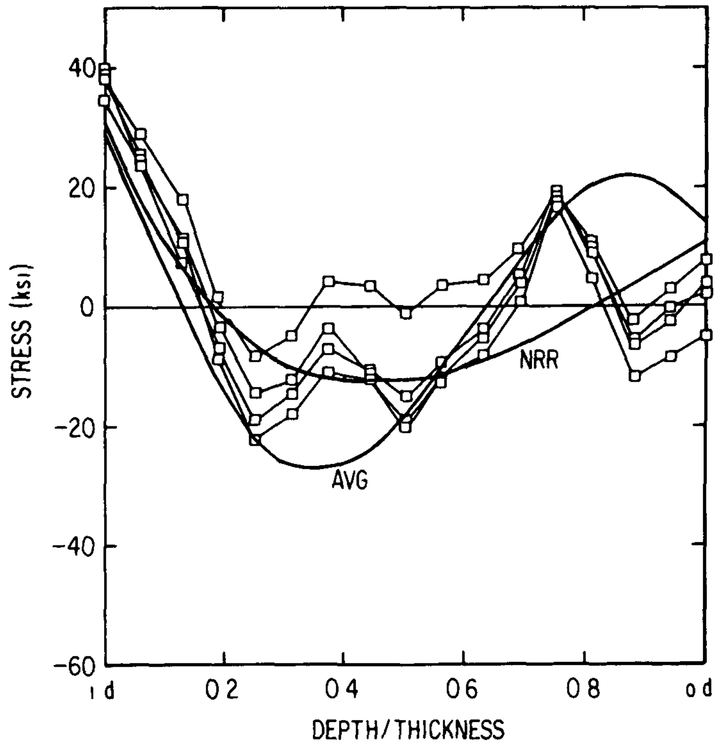


Figure 6.3 Throughwall distribution of axial residual stress in a large-diameter weldment.

calculations at least for semiquantitative prediction has been established. Although a number of finite element models for the welding process have been developed, most of the modeling work directly related to multipass welds on heavy-walled piping has been based on a finite element code developed at Battelle Columbus Laboratories (6.8) under NRC sponsorship and subsequently refined and modified under EPRI sponsorship (6.9).

The finite element calculations, which are supported by the limited experimental measurements available (6.11, 6.12), suggest that there are significant differences in the throughwall distributions of residual stress in small-diameter and large-diameter pipe weldments. Stress distributions in 10- to 12-in. weldments appear to be quite dependent on the weld heat input. The assumed stress distribution used for evaluation of flaws in small-diameter weldments (less than 12 inches) is shown in Figure 6.4. It is a conservative representation of the available data and computational results.

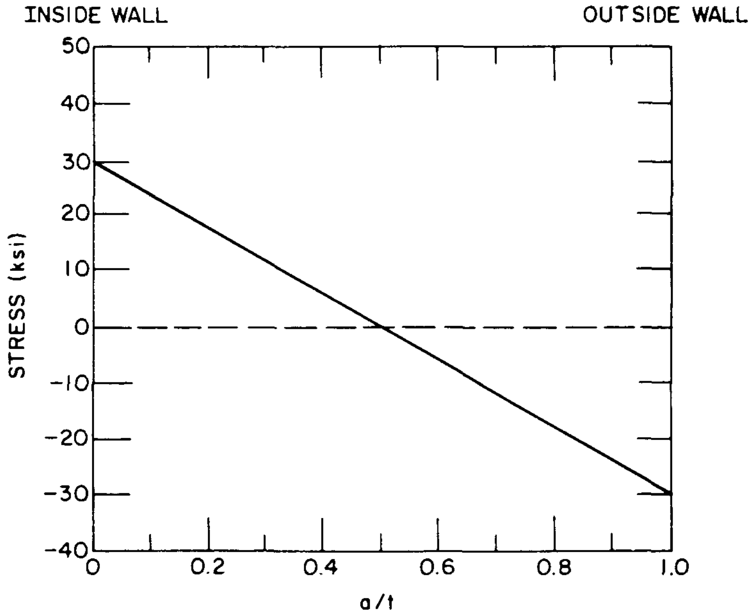


Figure 6.4 Assumed throughwall welding residual stress distribution in small-diameter weldments (<12 in.).

All of the available experimental measurements have been performed on pipe-to-pipe welds, which represent only a portion of the welds in a reactor piping system. However, similar results can be expected for pipe-to-elbow welds, since the local geometry near the weld is similar. Finite element calculations for a variety of weld geometries have been carried out (6.9). The results of these calculations indicate that the axial stress distributions used for evaluation appear to be very conservative for pipe to component welds.

Elastic Superposition and the Calculation of Stress Intensity Factors

The use of linear elastic fracture mechanics assumes that the residual stresses due to welding (or IHSI or a weld overlay) can be linearly superposed on the operating stresses to determine the total stress acting on the weldment. The assumption of linear superposition is compared with an elastic-plastic finite element analysis in Figures 6.5 to 6.7. For stress levels typically encountered in BWR piping systems the assumption gives satisfactory results, although it should be used with some caution with high applied stresses.

Stress intensity factors for circumferential cracks in piping can be calculated by a variety of methods. Probably the most convenient is the influence function technique (6.13, 6.14). For finite aspect ratio cracks, growth in both the throughwall and circumferential directions must be considered. However, the assumption of a complete circumferential crack, which is currently the basis for most assessments by NRR, give a conservative estimate of the stress intensity for throughwall crack growth (6.13).

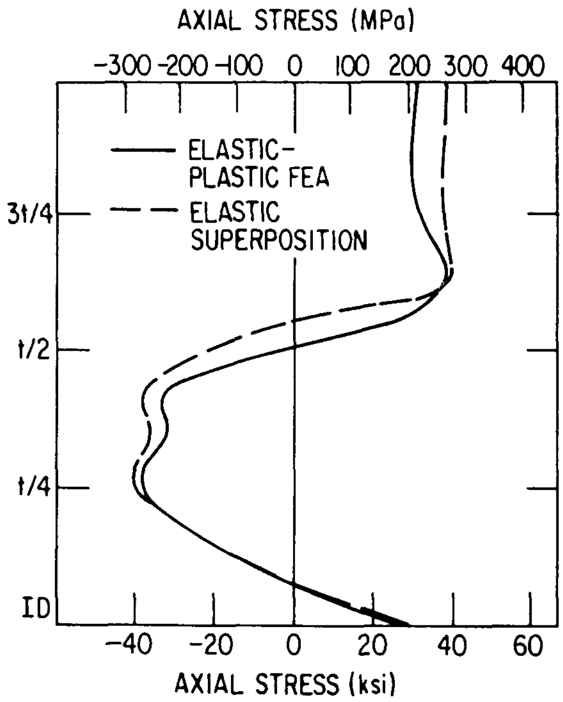


Figure 6.5 Comparison of elastic-plastic solutions for throughwall axial stresses with simple elastic superposition, 5-ksi applied load.

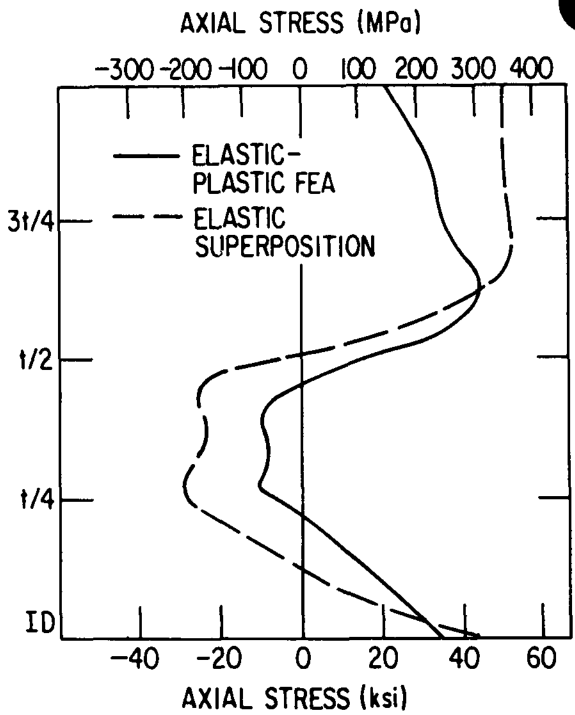


Figure 6.6 Comparison of elastic-plastic solutions for throughwall axial stresses with simple elastic superposition, 18-ksi applied load.

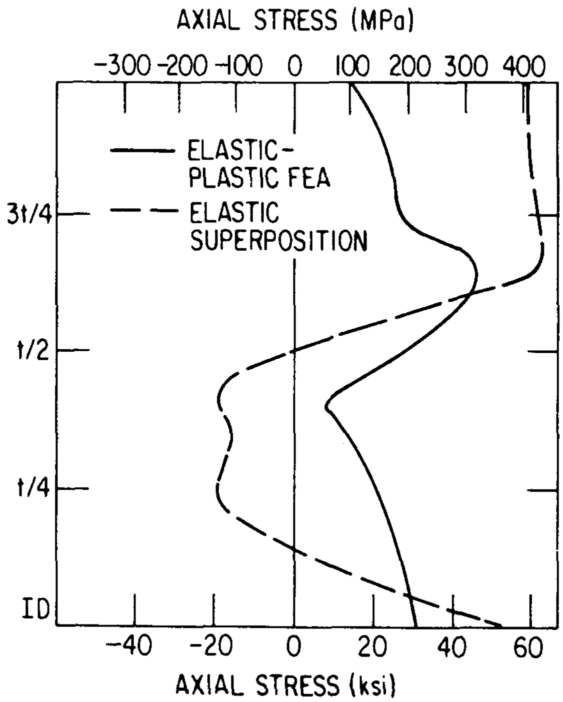


Figure 6.7 Comparison of elastic-plastic solutions for throughwall axial stresses with simple elastic superposition, 26-ksi applied load.

In addition to the assumption that the residual and operating stresses can be linearly superposed, it is also assumed that the stress intensity due to the residual stress can be calculated based on the residual stresses acting on the crack plane in an uncracked pipe. Clearly the residual stresses will redistribute and relax as the crack grows. The influence function solutions do account for this relaxation at least approximately. However, the validity of this approach in the presence of steep axial gradients of residual stress and in a region which has undergone extensive plastic deformation is still a somewhat open question for deep cracks (6.13) (i.e., over halfway through the wall).

6.2.1.2 Determination of Allowable End-of-Operation Crack Dimensions: Current Approach

To establish the acceptable crack sizes at the end of a specified operating interval, NRR allows only two-thirds of the crack depths provided in Paragraph IWB-3640 of Section XI of the ASME Code. The two-thirds allowable is intended to account for uncertainties in crack sizing and roughly translates to a factor of 2 on crack size. (If the original crack depth is actually twice the value reported from UT data, the crack will still remain within the IWB-3640 acceptable values at the end of 18 months of operation.) The description of the flaw acceptance criteria in SECY-83-267c further states that the staff criteria would likely require repair for cracks greater than 30% of the circumference in length and cracks with a reported depth 25% or greater of the wall thickness.

The acceptable flaw sizes presented in IWB-3640 are intended to provide a factor of safety against pipe failure of at least 2.77 for nominal operating conditions and at least 1.38 for emergency and faulted conditions. The critical crack dimensions in IWB-3640 were determined using net section collapse, also referred to as limit load or plastic collapse analysis. The applicability of net section collapse analysis to the evaluation of flaws in the wrought base metal of stainless steel piping was demonstrated in a series of pipe fracture experiments conducted at Battelle Columbus Laboratories (BCL) for the Electric Power Research Insitute (EPRI) (6.15) and the analytical studies reported in EPRI NP 2661.

The tests performed by BCL included 18 quasistatic experiments performed on Type 304 stainless steel at ambient temperatures. Eight of the experiments involved throughwall circumferential cracks in 2-, 4-, and 16-in.-diameter pipes. A total of ten circumferential surface crack experiments were performed in 4- and 16-in.-diameter pipe. All of the experiments involved bending loads. In an earlier program at BCL (6.16), two 4-in.-diameter throughwall crack tests were performed on stainless steel pipes under combined pressure and bending loads. The largest difference between the maximum load predicted by net section collapse analysis and the maximum load in the experiments was 14%, and the differences in the remaining tests were all within a few percent. The one exception was the test of a 16-in.-diameter surface flawed pipe. This experiment involved a surface crack approximately 66% throughwall and 170° around the pipe circumference. The maximum load observed in the experiment was approximately 14% below the failure load predicted by net section collapse. Overall, the tests conducted by BCL convincingly demonstrated the validity of net section collapse analyses under the range of crack dimensions, materials, and load combinations considered.

Recently, several concerns have been raised, from various sources, regarding IWB-3640 and the use of net section collapse analysis. These concerns can be summarized briefly as follows.

- Large uncertainties in the ability of ultrasonic testing techniques to size the depth of IGSC cracks accurately raise serious concerns about the ability to perform a reliable flaw evaluation.

- IWB 3640 is based on primary loads only and did not include secondary loads (e.g., thermal, support failures) in determining allowable flaw sizes.

- Flaw evaluation criteria should include consideration of "preservation of structural ductility" in piping systems with cracks present.

- Ductile fracture toughness data show much lower resistance to crack extension in stainless steel welds than in wrought stainless steel base metal. This lower toughness could invalidate the use of net section collapse analysis.

- The net section collapse formulation for circumferential cracks used in IWB-3640 can predict only gross failure of the cracked pipe cross section. There is a possibility that a surface crack could break through the remaining wall thickness before reaching net section collapse conditions and become a throughwall crack. This condition is referred to as a ligament instability.

These concerns are addressed in the following discussions.

Flaw Sizing Considerations

Section 4 of this report discusses the difficulties associated with the sizing of IGSC cracks. Results of tests conducted by Pacific Northwest Laboratory and EPRI suggest large deficiencies in the ability to measure the depth of IGSCC accurately. Furthermore, the trend in these round robins was to undersize deep cracks (greater than 20% throughwall). The current NRC flaw evaluation procedures address this issue by applying a factor of two-thirds to the flaw depths allowable by IWB-3640. In marginal cases, independent and additional crack depth measurements are required. Another proposal which has been made in an attempt to circumvent the problem of crack sizing is to establish an acceptance criterion based on crack length. The idea is that flaw length measurements of IGSCC are more reliable than depth measurements and a maximum acceptable crack length can be established such that even if the crack broke through the wall of the pipe it would be small enough in length that gross failure of the pipe would not occur and the resulting leak rate would be acceptably small.

Assumed Loads and Structural Ductility

The Task Group has the following opinions regarding the second and third issues described previously.

In general, secondary stresses should be included in the flaw evaluation if, as described in the next paragraph, the material toughness is low enough to preclude reaching limit load prior to crack instability. The Task Group's fracture mechanics analyses described in subsequent sections of this report include thermal expansion stresses, which are conservatively included as primary stresses. The Task Group further believes that other secondary stresses (e.g., through-the-thickness stresses) do not contribute significantly to crack driving potential and, in view of the conservative treatment of thermal expansion stresses, can be neglected in the flaw evaluation.

The secondary stresses resulting from support failures and the ductility issue apparently result from a concern that displacements during seismic events may be so large that loads on the supports and at locations in the piping system may exceed the ultimate load. It has been suggested that piping systems be shown to be able to tolerate added loads due to support failure and demonstrate that cracked sections have sufficient ductility (have net section plasticity) to absorb the energy associated with the postulated extreme displacements.

It is the Task Group's opinion that the conditions associated with the suggested extreme displacements do not represent a credible event. Furthermore, the suggested displacement and loading conditions are inconsistent with the ASME Code philosophy, which has been deemed acceptable for the design and operation of reactor components. Therefore, the Task Group has concluded that the loading conditions defined in the ASME Code are acceptable for performing evaluations of cracked piping. The suggested approach of postulating large displacements is acceptable but is considered overly conservative and should not be a general requirement.

Low Toughness and Ligament Instability Considerations

Regarding the issue of lower resistance to crack extension in weld deposits relative to wrought materials and the potential for ligament instability, the concern is that the assumptions associated with net section collapse may be invalid. Specifically, the assumption that no crack extension will occur prior to reaching maximum load may be incorrect. Destructive examination of a limited number of cracked weldments removed from service has revealed that cracks which initiate in the HAZ by SCC may grow into the weld metal. This type of cracking behavior also has been observed in laboratory tests. Although crack growth into welds is rare and, in service, has been associated with low-ferrite-content repair welds, the possibility that IGSC cracks may grow into the weld material, regardless of the ferrite level, cannot be ignored. In relation to IWB-3640, this issue must be addressed for both weld metal and cast stainless steel materials which also have reduced fracture toughness.

To address the issues of low fracture toughness and the potential for ligament instability, a series of fracture mechanics calculations using the J-integral tearing instability methodology was performed. Details of the analyses are presented in Appendix F. The assumptions made in the analyses and the results are summarized here.

The first set of calculations was performed for postulated elliptically shaped, inside-surface cracks half the pipe circumference in length and having assumed maximum crack depths of 25%, 50%, and 75% of the wall thickness.

These flaw geometries were evaluated for 12-, 22-, and 28-in.-diameter pipes with wall thicknesses typical of BWR recirculation system piping. The flaw geometry was selected because it is believed to be representative of actual stress-corrosion crack geometries and because this flaw geometry with the assumed 50% crack depth is believed to bound the most severe end of operating cycle crack geometry predicted for those cracked pipes which have gone back into operation with no repair.

Loading conditions evaluated include service level A (pressure, thermal, and deadweight loads) and service level A plus additional bending moments up to twice SSE load. These loading conditions correspond to stress ratios of 0.7 under normal operating conditions in 28- and 22-in.-diameter pipes and 0.9 in 12-in.-diameter pipes. The stress ratios for the service level A plus one SSE loads are 1.3 in 28- and 22-in.-diameter pipes and 1.5 in the 12-in.-diameter pipe. These loads represent reasonable upper bounds on actual plant loading conditions.

It is expected that the majority of the stress-corrosion cracks that occur in service will not lie solely in the weld metal but will be in zones of material having substantially higher resistance to crack extension (aside from IGSCC) than the weld metal. However, the potential for ductile extension of the surface cracks was evaluated for the above loading conditions using the lower bound of the ductile fracture toughness properties generated for the NRC for stainless steel welds. These data were generated from welds made using a hot wire tungsten inert gas welding process and are considered a lower bound of the fracture toughness properties for the majority of the circumferential welds in BWR piping. However, the Task Group has recently learned that a significant percentage (approximately 30%) of the circumferential welds in BWR piping are shop fabricated submerged arc welds. It has also been learned that some fraction of these welds, the exact number of which has not been tabulated, were not solution annealed. These findings are important because limited data show that ductile fracture toughness properties for submerged arc welds are significantly lower than those used in this evaluation. The Task Group is therefore initiating a high priority effort to evaluate the impact of the submerged arc weld fracture toughness data on the calculations presented here. The Task Group also recommends that NRC and industry initiate high priority efforts to generate more fracture toughness data for the range of weld types existing in BWR plants.

The results of the surface crack analyses are presented in Tables 6.1 and 6.2. Because of limitations in the available calculational techniques, analyses were not performed for cracks 75% of the wall thickness in depth at loads greater than service level A. These limitations in the analysis techniques did not allow a determination of the exact margins against fracture to be made. However, for the service level A normal operating condition loads, no ductile crack extension was predicted for crack depths up to 75% throughwall. At normal operating loads for the 50% deep crack (which is deeper than any known crack in service), the factors of safety (based on load) against ductile crack extension are at least 2.7 for the 22- and 28-in.-diameter pipes and 2.3 for the 12-in.-diameter pipe. The factors of safety against actual pipe failure will be larger than these factors of safety because there is additional load-carrying capacity beyond the initiation of crack extension to crack instability. A factor of safety against failure of

TABLE 6.1

Results of Fracture Mechanics Analyses for 22- and 28-Inch-Diameter Pipes With Surface Flaws 50% of the Circumference in Length

Loading Condition	Stress Ratio $\frac{P_M + P_B}{S_M}$	Maximum Crack Depth in % of Wall 25%	50%	75%
SLA	0.7	no initiation	no initiation	no initiation
SLA + 1 SSE	1.3	no initiation	no initiation	*
SLA + 2 SSE	1.9	no initiation	no initiation	*

TABLE 6.2

RESULTS OF SURFACE CRACK ANALYSES FOR 12-INCH-DIAMETER PIPE

Loading Condition	Stress Ratio $\frac{P_M + P_B}{S_M}$	Maximum Crack Depth in % of Wall 25%	50%	75%
SLA	0.9	no initiation	no initiation	no initiation
SLA + 1 SSE	1.5	no initiation	no initiation	*
SLA + 2 SSE	2.1	no initiation	no initiation	*

NOTES: SLA = Service Level A; SSE = Safe Shutdown Earthquake.
* = This case was beyond the limits of the analysis technique.

approximately 2.7 or greater would be predicted by IWB-3640, which is based net section collapse analysis for this crack geometry and normal operating condition loads. For the 50% deep crack under service level A plus 1 SSE loading the factors of safety are 1.5 for the 22-in.-diameter pipes and 1.4 for the 12-in.-diameter pipe. These factors of safety are approaching those intended in IWB-3640.

The results described in the previous paragraph were obtained for service level A and seismic stress conditions that represent the largest stress reported for the population of welds in 12-, 22-, and 28-in. BWR piping. These loads are associated with a relatively small percentage of the weld connections in the recirculation system. To illustrate the increased margins that exist for the majority of weld connections in the BWR recirculation system, surface flaw analyses were performed for a weld connection in a 28-inch pipe having a typical rather than bounding stress condition. The results from this analysis indicate that margins of approximately 3.3 for service level A conditions are obtained for most weld joints compared to values of 2.7 for the bounding stress location.

Because of the limitations in the surface crack analysis techniques, the factors of safety against ductile crack extension under emergency and faulted conditions cannot be determined for the 75% and deeper crack depths. Tearing instability calculations for throughwall cracks, presented in the next section on leak-before-break, give some insight into margins for these crack sizes and loading conditions.

6.2.2 Leak-Before-Break-Evaluations

The leak-before-break failure mode in piping refers to the concept that a crack propagating through the wall of a pipe, by mechanisms such as SCC or by ductile crack extension due to applied loads, will result in a stable through-wall crack that can be reliably detected by leakage. To evaluate these margins for leak-before-break, IGSCC analyses were performed assuming stress-corrosion cracks exist in the wrought stainless steel base metal. In addition, ductile fracture mechanics analyses were performed for cracks assumed to exist in the lower toughness stainless steel weld metal. These calculations are summarized below, and in Appendix F.

6.2.2.1 Margin for IGSCC Leak-Before-Break in Wrought Stainless Steel

The inherent toughness of Type 304 stainless steel is expected to lead to leak-before-break behavior, and to the knowledge of the Task Group, no sudden catastrophic failures of this material have occurred in the nuclear industry or in a wide variety of other industries. However, IGSC cracks in large-diameter pipe weldments are generally very long in relation to their depth; therefore, the possibility that cracking will occur 360° around the circumference before a throughwall crack develops must be considered.

Unless IGSCC growth is arrested by a favorable residual stress distribution, failure of the piping system could occur by continued growth of the crack by stress corrosion without the necessity for postulating loads on the piping beyond the normal operating and upset loads.

The determination of the margin for leak-before-break requires an analysis of the failure behavior of the pipe for a particular crack geometry, calculation of the crack opening area, and a calculation of the flow through the crack opening. In the results presented here it is assumed that the crack is not in the weld metal but lies in a zone of relatively high toughness material and the failure of the cracked pipe is assumed to be adequately described by a net section stress approach similar to that used in IWB-3640. This assumption is expected to be valid for the majority of the cracks that occur in operating BWRs. The calculation of the leak rate through the crack is subject to considerable uncertainty. The crack opening area has been estimated on the basis of elastic solutions for a throughwall crack in an infinite plate with an approximate correction for plasticity effects based on the Dugdale model (6.17). More complete elastic-plastic solutions for the crack opening area suggest that this solution is conservative (6.18).

The flow of steam-water mixtures through tight cracks is a complex function of crack geometry, crack surface roughness, temperature, and pressure. Limited experimental measurements of the fluid flux through IGSCC and simulated cracks have been carried out (6.19, 6.20). Comparisons of these data with the flow predicted by the homogeneous critical flow model developed by Henry (6.21) indicate that the actual flux is from 1-1/10 of the corresponding frictionless flux predicted by the Henry model. Although models which attempt to account for the frictional losses have been developed (6.6), the limited data base and the uncertainty about the internal crack geometry, corrosion product deposition, and roughness make it difficult to make accurate predictions of the flow through an IGSCC.

This uncertainty has surprisingly little effect on the assessment of the leak-before-break margin in piping. Figure 6.8 shows a typical case for a 10-in. pipe. It is assumed that the crack can grow completely around the pipe circumference before a throughwall crack occurs. The collapse curve in Figure 6.8 assumes that the net section stress at failure equals 3 S_M. The observed variations in flux are used to estimate upper and lower bounds on the crack sizes necessary to obtain a 5-gpm leak rate. Based on crack size there is a significant leak-before-break margin unless the circumferential crack is quite deep. Because of the steep nature of the collapse curve for deep circumferential cracks, the size of the crack that can be postulated before violation of leak-before-break is not very dependent on the value chosen for the fluid flux (6.22).

The corresponding case for a 24-in. pipe is shown in Figure 6.9. The size of the crack needed to obtain detectable leakage is only very weakly dependent on pipe diameter, while the size of the crack needed to produce collapse is roughly proportional to pipe diameter. Hence the margin for leak-before-break generally increases with pipe size. As Figures 6.10 and 6.11 show, however, the leak-before-break margins are smaller at highly stressed joints.

Estimates of the time from the onset of unallowable leakage to structural failure depend strongly on the mechanisms of crack growth and the corresponding crack growth rate assumptions. In most cases most of the remaining crack growth will occur by stress corrosion. Figure 6.12 summarizes the available data on IGSCC growth rates in BWR type environments. Both data from

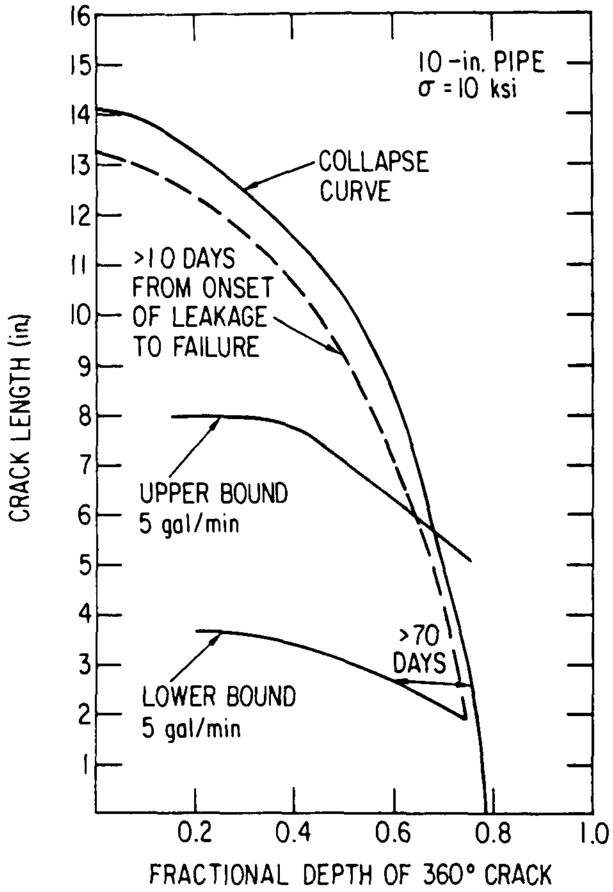


Figure 6.8 Collapse curve for a 10-inch pipe with throughwall and part-through 360 cracks with bounds for crack sizes for 5 gpm. Leak rate under an applied stress of 10-ksi.

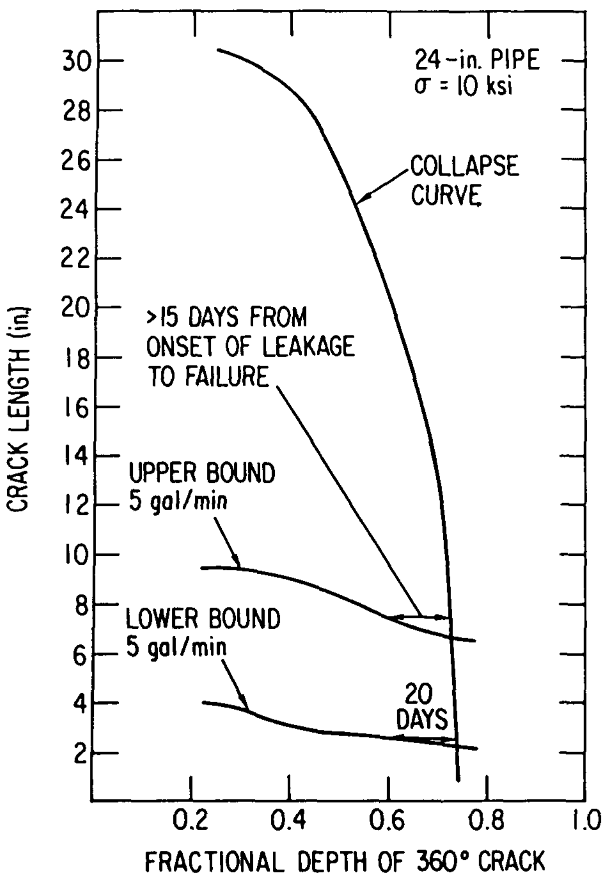


Figure 6.9 Collapse curve for a 24-inch pipe with throughwall and part-through 360 cracks with bounds for crack sizes for 5 gpm. Leak rat under an applied stress of 10-ksi.

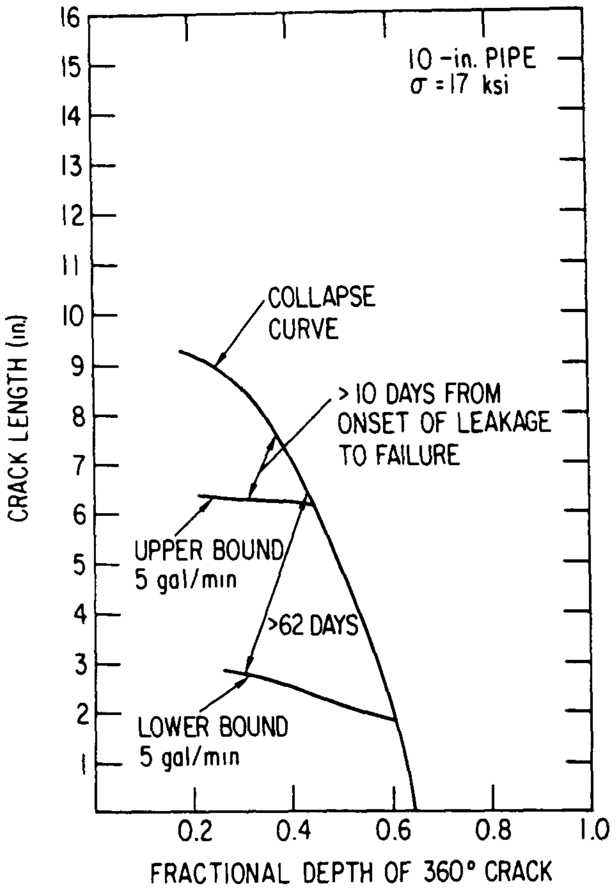


Figure 6.10 Collapse curve for a 10-inch pipe with throughwall and part-through 360 cracks with bounds for crack sizes for 5 gpm. Leak rate under an applied stress of 17-ksi.

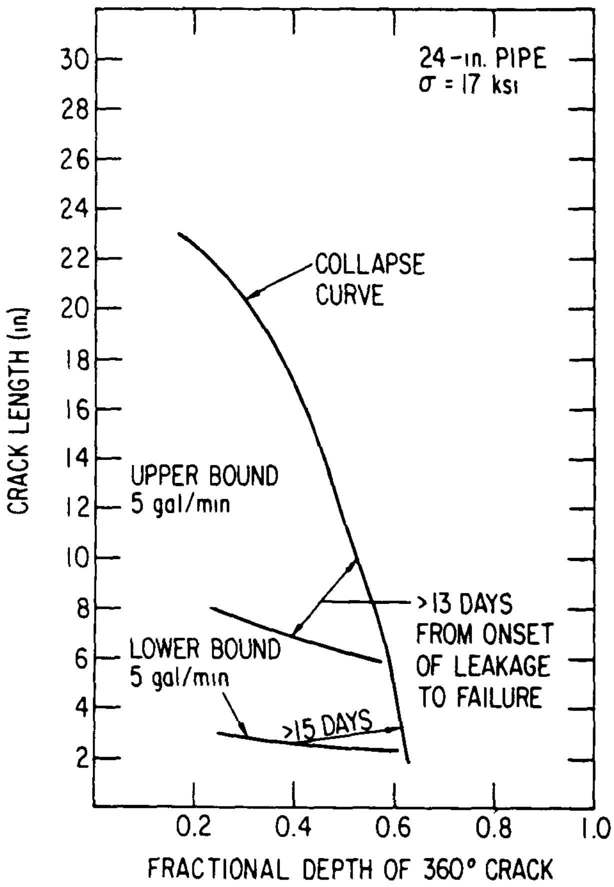


Figure 6.11 Collapse curve for a 24-inch pipe with throughwall and part-through 360 cracks with bounds for crack sizes for 5 gpm. Leak rate under an applied stress of 17-ksi.

fracture mechanics tests under controlled load and slow strain rate tests in which the load is varied to maintain the nominal strain rate constant are included. It is unlikely that the strain rates commonly used in slow strain tests can be maintained under reactor loading conditions, and therefore an upper bound for in-reactor loading based on growth rates in slow strain tests should be very conservative. The crack growth curve selected in Figure 6.12 bounds all available fracture mechanics data (the plateau rate is ten times the highest reported rate in a constant load fracture mechanics test), and the plateau rate is typical of slow strain rate tests.

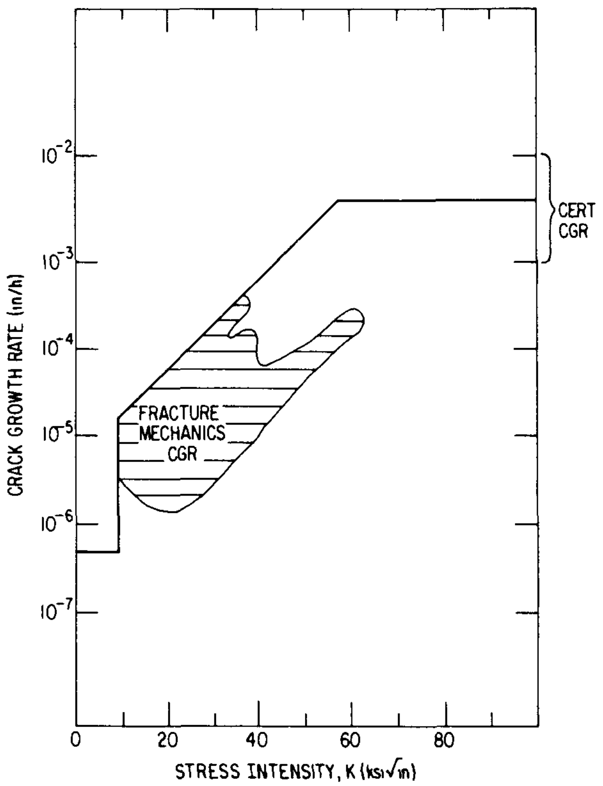


Figure 6.12 Conservative estimate of upper-bound stress-corrosion crack growth rates of Type 304 stainless steel in BWR environments.

Using this crack growth rate curve and using the usual linear elastic fracture mechanics approach, the time from the onset of leakage to failure can be estimated for different crack geometries. For simple throughwall flaws conservative estimates of the times range from two months for the 10-inch pipe to three months for the 24-in. pipe. Even for the case of the compound crack with a complete circumferential part-through crack in addition to the throughwall crack, the time is relatively long for most crack geometries. Figure 6.8 shows the bounding envelope of crack geometries for which there are at least ten days before the onset of unallowable leakage and failure. Leak-before-break in this sense is violated only in the case of very deep, very long part-through cracks.

Leakage from cracks that are not large enough to cause collapse can be very large. For a 24-in. pipe, the leak rate through a 22-in. crack (approximately 30% of the circumference) under typical applied stresses is approximately 100 gpm, which is close to the makeup capacity of most BWRs. Hence, while leak-before-break is still valid, in many cases with cracks greater than 30% of circumference, the resulting leakage could require use of the ECCS.

6.2.2.2 Margins for Leak-Before-Break in Stainless Steel Weld Metal

For cracks located in the stainless steel weld metal, the lower toughness relative to the wrought material may invalidate the use of limit load analysis. For this reason a series of tearing instability calculations were performed for simple throughwall crack geometries. The objective of these calculations were twofold. First, to determine if throughwall cracks of a length reliably detectable by leakage have adequate margins against fracture and, second, to determine the maximum length of crack that could become throughwall and remain stable.

For these analyses the same loading conditions and material properties used for the tearing instability analyses in the section on flaw evaluation procedures were used. The crack geometry considered is a simple throughwall crack. Details of these calculations are presented in Appendix F. The loads for crack instability were calculated for throughwall crack lengths corresponding to a 10-gpm leak rate under normal operating conditions (service level A) in 12-, 22-, and 28-in.-diameter pipes. These crack lengths were 7.2 in., 9.6 in., and 10 in., respectively. The 10-gpm leak rate represents a safety factor of 2 relative to the 5-gpm limit for unidentified leakage currently imposed on most operating BWRs.

The results of the calculations showed that under normal operating conditions (service level A) the factor of safety on stress against unstable fracture of the throughwall cracks ranges from a minimum of 2.5 for the 12-in.-diameter pipe to approximately 3.5 for the 22- and 28-in.-diameter pipes. For faulted conditions (service level A plus one SSE) the factors of safety are approximately 1.5 for the 12-in.-diameter pipe and 1.9 for the 22-and 28-in.-diameter pipes. Thus, for 12- to 28-in.-diameter pipes significant margins against unstable fracture exist for cracks which should be reliably detected by leakage.

In addition to the above, calculations were performed to determine approximately what length of throughwall crack would become unstable under the assumed bounding loading conditions (service level A plus 1 SSE). The calculations are important for two reasons. First, they show the large difference between the throughwall crack length that should be reliably detected by leakage and the throughwall crack size that could result in unstable fracture of the pipe. This difference in crack length indicates that there is a large amount of time in which to detect leakage from a throughwall crack before it reaches a critical crack length, thus providing additional confidence in the applicability of leak-before-break.

The calculations are also important because of the large uncertainties associated with the ability to size the depth of IGSC cracks (see Section 4). These uncertainties suggest that it would be desirable as a flaw evaluation procedure to limit the length of acceptable IGSC cracks such that if they became throughwall cracks under normal or postulated accident conditions they would not result in unstable pipe fracture or unacceptably large leak rates. The limit on leak rates stems from the desire not to challenge emergency core cooling systems.

Tearing instability calculations were performed for throughwall cracks extending around 30% and 26% of the circumference for 22- and 28-in.-diameter pipes, respectively. The crack length resulted in factors of safety on stress of approximately 1.2 for the two pipe diameters under faulted loading conditions. These calculations indicate that even in the event that a surface crack, of the type evaluated in the section on flaw evaluation, suddenly became a throughwall crack under accident loading conditions, it would remain stable if the crack were less than approximately 30% of the circumference in length. On a weld joint specific basis the length of the critical throughwall crack based on fracture mechanics would be larger than 30%, due to lower applied stresses than assumed in the analysis.

6.3 SHORT-TERM SOLUTIONS

This subsection is concerned with measures considered acceptable by the NRC for relatively limited periods; e.g., 1 to 2 fuel cycles. In most instances weld overlays or similar approaches are used to "buy time" where the assumption is that many of the piping systems containing several cracks may ultimately have permanent corrective measures. Measures such as IHSI described in subsection 6.4.3 may represent a short-to-long-term mitigative measure. The event tree approach in Section 6.1 attempts to display the interaction of the various factors impacting on a decision to continue to operate or to make limited or longer-term repairs.

A variety of techniques have been used or proposed for short-term repair of flawed piping including replacement of a short portion of the piping, a welded "clamshell," mechanical clamping devices, and reinforcement by deposit of weld metal on the outer surface of the weldment (weld overlay). In this country the vast majority of repairs have been done by weld overlay. The weld overlay obviously provides structural reinforcement; in addition the radial shrinkage induced by the overlay produces strongly compressive residual stresses on the inner surface of the weldment, and the low-carbon, high-ferrite level, Type 308L weld metal used for the overlays is more resistant to IGSCC than sensitized Type 304 stainless steel.

The compressive stresses induced by the overlay should be effective in preventing the initiation of new intergranular stress-corrosion cracks and inhibiting the growth of existing flaws. Calculations suggest that for crack depths up to 60% of the wall thickness the stresses ahead of the crack will be compressive for applied stresses up to 9 ksi, which is typical of the stress levels in large-diameter piping. However, because of uncertainties in crack depth measurement, it is difficult to assure that any particular crack is under a favorable stress condition that would inhibit further IGSCC growth. At present it is also not possible to rely on the weld overlay itself to arrest IGSC crack growth, since although it has been clearly demonstrated that this type of material is very resistant to crack initiation, its ability to arrest crack growth in the presence of high stress intensity factors has not been convincingly demonstrated. Since it is also very difficult to carry out reliable inservice inspections through the overlay, the Task Group concurs with the current NRR position that the weld overlay be considered as a short-term remedy used primarily to allow time to prepare for the replacement of severely cracked pipe. In cases in which the weld overlay is applied to relatively short deep cracks to prevent leakage rather than to prevent structural failure, longer-term operation with weld overlays can be considered. Because of the resultant difficulties in inspection, routine application of weld overlays to induce favorable residual stresses in weldments with no detectable cracking as an alternative to IHSI is not recommended.

6.4 LONG-TERM SOLUTIONS AND SAFETY ASSESSMENT

6.4.1 Introduction

The proposed long-term remedy procedures for pipe cracking in BWRs can be classified into three groups: alternative materials, residual stress improvement remedies, and alternative water chemistries. The alternative materials that have been considered include both new piping materials such as nuclear grade Types 304, 316 and 347, Types 304L and 316L, Types 347, CF3, and Types 304LN and 316LN stainless steels and remedies such as corrosion-resistant cladding which introduce only a localized region of resistant material in the vicinity of the heat-affected zone. Residual stress improvement remedies such as Induction Heating Stress Improvement (IHSI), Last-Pass Heat Sink Welding (LPHSW), and Heat Sink Welding (HSW) are intended to produce compressive residual stresses (or at least lower tensile stresses) on the inner surface of a weldment. The most promising alternative water chemistry utilizes hydrogen additions to feedwater to reduce the dissolved oxygen produced by radiolysis in the core. Reactor experiments in both the U.S. and Sweden have indicated that lower oxygen levels are achieved, and laboratory studies indicate that the lower oxygen levels can reduce or even eliminate susceptibility to IGSCC.

6.4.2 Alternative Materials

The alternative materials which have been considered for BWR piping systems include low-carbon stainless steels, stabilized stainless steels, higher chromium steels such as the Nitronic alloys, and cast stainless steels with duplex austenitic-ferritic structures. For a variety of reasons [ease of fabrication, familiarity, ASME Code acceptability, ease of inspection, etc. (6.23)] most attention has focused on low-carbon stainless steels.

The regulatory position on the actual procedures for the replacement of piping is outlined in Appendix C-2.

Low-carbon stainless steels such as Types 304L and 316L have long been widely used in a variety of industrial applications to avoid sensitization and the associated cracking problems. However, because of their relatively low carbon levels (less than 0.03%), they have lower design allowable stresses than conventional Types 304 and 316 stainless steels and hence cannot in general be used as a one-to-one replacement for the higher carbon stainless steels.

The nuclear grade stainless steels have even lower maximum allowable carbon levels (less than 0.02%), but nitrogen levels are also controlled within specified levels (0.06% to 0.10% N) to ensure that materials have strength levels that meet the ASME Code requirements for conventional stainless steels, and thus they can be used as a one-to-one replacement for conventional stainless steels without reanalysis or redesign of the piping systems (at least for piping systems which were designed to meet ASME Code requirements). It should be noted that the nitrogen levels for the nuclear grade steels are within the Code specifications for Types 304 and 316 stainless steels (N less than 0.1%). The LN-grade steels are nitrogen-strengthened steels somewhat similar to the nuclear grade steels, although the allowable carbon and nitrogen limits are slightly different (less than 0.035% C and 0.06% to 0.16% N). They have recently been recognized by the ASME Code as acceptable materials for primary pressure boundaries.

Both Types 304 and 316 nuclear grade materials have performed well in a wide range of laboratory tests including full-scale pipe tests, slow strain rate tests, simulated crevice corrosion tests, and cyclic crack growth tests carried out in the U.S. and Japan (6.24, 6.25). Some of the data suggest that low levels of molybdenum play an important role in the stress-corrosion cracking resistance of these materials (6.27). The higher nickel and the molybdenum additions in Type 316 also make it much less susceptible to martensite formation during cold work than Type 304 (6.28). Thus, use of Type 316 nuclear grade material seems to provide additional protection against IGSCC produced in sensitized regions and cracking associated with cold work, and it appears to be the material which will be used by most utilities in replacement programs and for new piping.

The major laboratory testing program to qualify the nuclear grade stainless steels has been the pipe testing program carried out by General Electric under EPRI and BWR Owners Group (BWROG) sponsorship. The pipe test specimens were fabricated by butt welding eleven 100-mm-long segments of 4-in. Schedule 80 pipe together to form a 1.1-m-long test section. Transition pieces and end caps were welded to the test section. During the test the specimens were cyclically loaded while high purity water at 550°F (288°C) containing approximately 6 ppm dissolved oxygen was circulated through the test section. The conductivity of the water was maintained below 1 micro siemens per centimeter, but detailed records of the conductivity levels during the tests and the nature of the impurities are not available.

The load in these tests was increased linearly over a 5-min period until the stress reached a value of approximately 34 ksi, held at that level for 75 min, reduced linearly to a value of approximately 4 ksi over a period of 0.5 min, and held there for 5 min; the cycle was then repeated. For conventional stainless steels with 0.06-0.07 C contents, these tests produced throughwall intergranular cracks in approximately 70 cycles (6.29). No intergranular failures were observed in corresponding tests on 9 heats of Type 316NG and 8 heats of Type 304NG stainless steels in tests ranging from 2000-6000 cycles (6.30). One Type 316NG specimen did fail after 2885 cycles. This crack was reported to be outside the weld heat-affected zone near the weld counterbore and primarily transgranular in nature (6.30). A similar failure was observed in a Type 316L (0.024 C) specimen after 1256 cycles. Because of the character of the failures, these specimens were considered test anomalies by General Electric. Even including these particular cases, the materials show much superior performance in the pipe tests.

Extensive constant load, U-bend, and slow strain rate tests on five heats of Type 316NG in high purity, high temperature, oxygenated water have also been performed by the Japanese (6.31). The specimens in these tests were given a wide variety of heat treatments and surface treatments, including grinding. No stress-corrosion cracking was observed in these tests. Creviced bent beam tests (with graphite wool in the crevice) and double-U-bend tests (with a Teflon film between the two U-bends to form a tight crevice) were carried out to evaluate the susceptibility of these materials to crevice corrosion. Some intergranular cracking was observed in specimens subjected to furnace heat treatments [1100-1400°F (600-750°C)/30 h], but none was observed in welded specimens, even with an additional low temperature heat treatment [932°F (500°C)/24 h].

Hot tear tests, hot ductility tests, and varestraint tests (i.e., bend tests during welding) were also carried out to evaluate the weldability of Type 316NG stainless steel (6.31). These tests indicated that the weldability of the nuclear grade material was equivalent or in some cases superior to that of conventional Type 316 stainless steels. The lack of any adverse effect of nitrogen additions on weldability has been noted by other investigators (6.32). Recently, shallow (10-20 mil), intergranular cracking has been observed near seam welds in rolled and welded Type 316NG piping. The problem appears to be confined to one fabricator. The current hypothesis proposed by EPRI (6.33) is that the problem is due to copper contamination produced by the use of air-arc cutting with copper-coated carbon electrodes during fabrication of the pipe. During the seam welding process the copper oxide left on the surface during the cutting is reduced and the molten copper metal produces some grain boundary cracking. The explanation seems plausible and consistent with the absence of the problem at another fabricator using mechanical cutting, but additional study is needed to resolve the problem.

Cracking of Type 316NG during hot forming has been observed in some cases. This has been attributed to nonuniform heating and improper temperatures during the forming operation. Nitrogen-strengthened stainless steels have been widely used in Europe for many years and seem to have good fabrication and welding properties. Japanese tests and experience are consistent with this. However, care should be taken to ensure that the materials (in terms of impurity content etc.) and fabrication procedures used in this country are comparable to those used in Europe and Japan so that this experience is truly relevant.

The qualification tests performed by General Electric also included corrosion fatigue tests. Cantilever beam fatigue initiation tests were performed on welded specimens in high purity water with approximately 6 ppm dissolved oxygen at 550°F (288°C) using fully reversed loading at approximately 4 cph. One heat of Type 316NG (7 specimens), one heat of conventional Type 316 (7 specimens), and two heats of conventional Type 304 (6 specimens) stainless steels were tested. The Type 304 stainless steels failed intergranularly: The Type 316NG and Type 316 materials failed transgranularly. At 1.3% strain the life of the nuclear grade material was approximately four times that of the conventional materials, and at 0.75% cyclic strain it was approximately twice the life of the conventional Type 316 material and approximately four times the life of conventional Type 304 material. Cyclic crack growth rate tests were carried out on one specimen each of Types 316NG, 347, and 304 stainless steels using a trapezoidal waveshape at a cyclic frequency of 4.4 cph and a load ratio R = 0. The crack morphologies observed were transgranular in all cases. The crack growth rates for the Types 316NG and 347 materials were approximately 40% less than for the Type 304 stainless steel.

Intergranular crack growth has been observed in Type 316NG stainless steel in laboratory tests on precracked specimens (6.34). In tests on bolt-loaded fracture mechanics-type specimens with fatigue precracks, intergranular crack propagation was observed in a variety of alloys including Types 347, 316L, 316NG, 304L, XM-19 (Nitronic 50), and 304 stainless steels. The tests were carried out in high purity water (i.e, conductivity less than 1.5 micro siemens per centimeter, but no deliberate additions of impurities and no detailed history of the actual water chemistry) with approximately 6 ppm dissolved oxygen at 288^{o}C. The nominal initial stress intensity factors (K) were quite high in these tests, ranging from 40-70 ksi-in.$^{1/2}$. The final measured stress intensities were lower (20-55 ksi-in.$^{1/2}$) presumably because of the load relaxation produced by crack growth and time-dependent plasticity at the crack tips of the specimens. Microstructural, sensitization (EPR), and slow strain rate tests indicated no sensitization (carbide precipitation) of the nuclear grade steels. However, fully intergranular cracking was observed on the specimen fracture surfaces. General Electric has postulated that the presence of the intergranular cracking is attributable to the crevice crack-tip chemistry as well as to the extremely slow strain rate in the crack propagation process.

The higher nitrogen levels in both the nuclear grade and the LN materials raise the possibility that these materials may be more susceptible to transgranular cracking, since deleterious effects of nitrogen on the transgranular chloride cracking of austenitic stainless steels (primarily in $MgCl_2$ environments) were observed previously (6.37), although for materials close to the composition of the nuclear grade stainless steels the available results are somewhat ambiguous and little work is available in BWR-type environments.

The behavior of Type 316NG material in environments containing chlorides and sulfates has been considered in pipe tests by General Electric and in slow strain rate tests at Argonne National Laboratory (6.35). The water chemistries in the pipe tests generally simulated high impurity levels that are well beyond the normal range of BWR water chemistries and are characteristic of events like the Millstone chloride intrusion or the 1974 resin intrusion at Duane Arnold. The limited number and duration of the pipe tests make it difficult to draw conclusions from this work. In all cases the nuclear grade materials performed as well as the reference Type 304 materials, but in most cases the tests were ended without failures in either the nuclear grade or the conventional materials so that it is difficult to assess the relative performance of the two types of materials.

The tests at Argonne have focused on water chemistries with much lower levels of impurities that would be more typical of usual reactor operation. In no case has intergranular cracking been observed in nuclear grade materials even after fairly severe furnace heat treatments [122°F (650°C)/24 h) or in weldments given an additional low temperature heat treatment [932°F (500°C)/24 h]. However, transgranular stress-corrosion cracking has been observed in slow strain rate tests (6.36) in water chemistries with 0.2 ppm dissolved oxygen and with sulfate and chloride impurity levels consistent with the BWR water chemistry limits outlined in Reg. Guide 1.56 (approximately 0.1 ppm sulfate or chloride added as acid). For the same weight concentrations sulfate additions appear to be somewhat more detrimental than chloride additions.

In environments in which the nuclear grade materials exhibit transgranular cracking, sensitized Types 304 and 316 stainless steels exhibit intergranular cracking. Depending on the environmental conditions the transgranular crack growth rates in the nuclear grade steels are roughly factors of 3-10 lower than the intergranular crack growth rates in the conventional steels. The transgranular cracking appears to be associated with the impurity additions and is not observed in tests in high purity water. However, the minimum impurity levels required to produce cracking are not known at this time. The relative susceptibility of unsensitized conventional austenitic steels (which seem to have exhibited satisfactory resistance to transgranular cracking in-reactor) compared to the nuclear grade steels is also unknown at this time. It is also possible that the transgranular cracking observed in the slow strain rate tests requires unrealistically large plastic strains for initiation. This too is an area which requires additional testing.

It is well known (6.38, 6.39) that long-term exposure of Type 316 stainless steel at elevated temperatures can cause decomposition of the austenitic matrix into various intermetallic phases (e.g., sigma, laves, chi) that can have a very detrimental effect on the mechanical properties. The available data (6.39) on conventional Type 316 stainless steel suggest that this is unlikely to occur at BWR operating temperatures. Lower carbon levels seem to promote the formation of the intermetallic phases (6.38). If, as seems likely, this is due to the role of carbon as an austenite stabilizer, then the higher nitrogen levels of the nuclear grade may be beneficial, since nitrogen is also a strong austenite stabilizer [e.g., nitrogen additions appear to suppress the formation of martensite during cold working of 18/8 stainless steels (6.32)]. Thus, it seems unlikely that long-term service at BWR operating temperatures will cause significant degradation of the mechanical properties of the Type 316NG material.

Overall the nuclear grade materials appear to be a much better choice for BWR piping systems than the conventional austenitic stainless steels, but careful control and monitoring of water chemistry may be necessary to ensure total protection from both intergranular and transgranular stress-corrosion cracking. The low-carbon austenitic stainless steels have performed very satisfactorily in the EPRI/GE pipe tests and have performed well in-reactor in some repair situations. Similar performance is expected from the LN grades, since nitrogen additions do not appear to affect susceptibility to sensitization adversely (6.31, 6.32). However, as noted for the nuclear grade materials, nitrogen additions have been observed to increase susceptibility to transgranular cracking in some systems, and recent laboratory studies have shown that small levels of impurities greatly increase the relative susceptibility of very lightly sensitized materials (6.35). This suggests that the performance of the L grade materials (with C greater than 0.02%) may not be satisfactory in an actual reactor environment as the laboratory tests in high purity environments would indicate, and the lower carbon nuclear grade materials offer an additional margin against cracking that may be significant in actual reactor environments. Similarly, it seems prudent to restrict nitrogen additions to levels as low as possible while providing adequate mechanical properties.

Corrosion-resistant cladding in which the region in the vicinity of the weld is clad with duplex Type 308L weld metal with ferrite levels greater than or equal to 8% also appears to be a reasonably satisfactory remedy (6.40). In many cases it is not possible to solution-heat-treat piping after the cladding is applied so that a sensitized region remains on the inner surface of the weldment at the end of clad region. However, this sensitized region is limited in depth into the pipe wall. Hence, although the procedure is unlikely to produce complete immunity to cracking, it does significantly reduce the amount of sensitized material exposed to the environment. It also complicates the ultrasonic inspection of the weldment, and good baseline data are essential for satisfactory resolution of the ultrasonic reflections associated with the weld clad layer from possible cracking indications. Because of the inspection difficulties and the possible existence of sensitized regions on the inner surface, use of an alternative piping material is preferable.

6.4.3 Residual Stress Improvement Remedies

A number of residual stress improvement remedies have been proposed including Induction Heating Stress Improvement (IHSI), Last Pass Heat Sink Welding (LPHSW), and Heat Sink Welding (HSW). These procedures are intended to produce compressive residual stresses (or at least lower tensile stresses) in the inner surface of a weldment. (HSW may also provide an additional benefit by reducing the degree of sensitization near the weldment, although since the weldment is "seeded" with carbides, low temperature sensitization can significantly reduce the benefit associated with the initially lower degree of sensitization.) For operating plants, HSW is typically useful only in repair situations to reduce the susceptibility to cracking in the welds joining the new replacement piping to the original susceptible material. IHSI and LPHSW can be applied to weldments in situ. IHSI has been widely used on existing plants in Japan.

It has been demonstrated that both IHSI and LPHSW produce highly compressive residual stresses on the inner surface of butt piping weldments (6.41 - 6.43). Analytical studies indicate, however, that under applied loads of the magnitude expected in-reactor the total stresses on the inner surface may be tensile (6.44) but that in most cases the total stresses will be much less tensile than in a conventional weldment. Because the stresses may not be compressive, IHSI may not completely eliminate the possibility of IGSCC, but the lower stresses should mitigate the problem.

Experimental verification that the compressive residual stresses produced by IHSI are actually effective in inhibiting stress-corrosion cracking has been provided by pipe tests on 4-in. welded pipes carried out by General Electric under EPRI/BWROG sponsorship. The pipe specimens, test environment, and loading histories were similar to those described in Section 6.4.2 for the alternative materials. To prevent relaxation of the residual stresses induced by the IHSI, somewhat lower applied stresses (approximately 30 ksi) were used in these tests. These stresses are still substantially higher than would be found in most reactor operating situations, and would be expected to reduce significantly the compressive residual stresses induced by the IHSI process, and hence the expected benefit. However, pipes without preexisting flaws that were treated by IHSI still showed significantly longer lives than comparison pipes without IHSI (6.41). At applied stresses of 28.1 and 30.6 ksi, IHSI-treated pipes failed at lives ranging from 573 to 991 cycles. At higher stresses IHSI-treated pipes showed no improvement, which is consistent with the idea that the improvement is due to the favorable residual stresses.

Tests on pipes with initial defects showed no significant benefit of the IHSI process even for relatively shallow (less than 10% throughwall) pre-cracks (6.44). General Electric attributes this to the relatively high loads in the pipe tests, and they argue that with actual plant loads a significant benefit is likely for the shallow cracks that might escape detection. Their conclusions are consistent with available finite element results. Detailed analytical calculations (6.45, 6.52) suggest that at least for shallow cracks (less than 10% throughwall) the stresses ahead of the crack tips are compressive under most applied loads. Even for 40% throughwall cracks, the stresses ahead of the crack are compressive under typical applied loads, although they become tensile for applied stresses approximately greater than 10 ksi. The Japanese have treated precracked weldments with IHSI and then immersed them in boiling $MgCl_2$ solution (6.42). No additional crack extension was observed, which suggests that the stresses ahead of the crack tip were compressive or at least very low. However, this test does not address the possibility of the redistribution of stresses ahead of the crack under additional applied loads.

The Japanese have applied IHSI to piping with UT indications (6.53) in an operating reactor. After a year's operation, there was no apparent crack growth. However, the indicated flaws were very small (less than an inch in length).

Although the finite element results and the limited Japanese experience are encouraging, uncertainities in UT sizing and the stress levels at particular joints require that the effectiveness of IHSI on flawed joints be evaluated conservatively. However, the evidence indicates that IHSI should be effective for small flaws that have a relatively low probability of detection and, in fact, it offers some improvement even for somewhat larger flaws.

For precracked pipes there is also the possibility that cracks may be extended mechanically during the IHSI process (although the inner surface of the weldment is in compression after completion of the process, it undergoes a tensile loading during the process). Metallographic studies on precracked weldments subjected to IHSI by the Japanese (6.42) and General Electric (6.41) suggest this does not occur at least for relatively shallow cracks (less than 25% throughwall). Finite element studies (6.52) indicate that deeper cracks will not propagate mechanically during IHSI (the calculations were done for 40% throughwall cracks).

Both analytical results (6.44) and the pipe test results (6.41) suggest that under the primary loads expected in piping systems the stresses on the inner surface of weldments treated by IHSI will be lower than for nontreated welds, which should have a mitigating effect on IGSCC. However, since only limited amounts of plastic strain are necessary to relieve the favorable residual stresses introduced by IHSI, it is probably also prudent to consider the possibility that secondary and peak stresses which produce small localized plastic strains may also relax the stresses induced by IHSI. A comparison of plastic strain levels suggests that local plastic strains of the order of those produced globally by primary stresses in the finite-element analyses can be expected in regions with high secondary and peak stresses. Thus it is possible that the IHSI-induced stresses will be relaxed in these regions. However, in such cases the high tensile residual stresses produced by welding will also be relieved, and the situation at least from the standpoint of the stresses at the weld, will be no worse after IHSI.

In addition to relaxation under applied loads, it is also possible that the stresses induced by IHSI will be relaxed by creep deformation. However, Japanese studies (6.43) indicate that very little relaxation will occur at the operating temperature of a BWR during the 40-year design life. The available limited measurements on residual stresses in weldments from operating reactors (6.46) also suggest that little relaxation occurs at these temperatures.

The effect of the critical process parameters (coil length, heating rate, peak temperature of the outer surface, etc.) on the residual stresses produced by IHSI has been extensively studied both in the U.S. and Japan (6.41-6.43, 6.47). The effectiveness of the process parameters currently recommended by the process vendors in producing high compressive residual stresses on the inner surface of piping has been convincingly demonstrated for pipe-to-pipe butt weldments (6.41, 6.42). Much less data are available for pipe-to-component weldments. Japanese data suggest that the process is less effective in inducing compressive axial residual stresses in these cases (6.42). However, these data were obtained using an IHSI procedure which differs in some respects from the current procedures recommended by the IHSI vendors, and recent experimental and analytical results suggests that significant compressive stresses can be obtained, at least for some pipe-to-component configurations (6.41, 6.47). It should also be noted that the data do indicate a substantial reduction in the circumferential stresses, which lead to axial cracks, for the component configurations, and in no case do the data indicate that the IHSI process produces a less favorable residual stress state than the as-welded condition.

During the IHSI process, the treated weldment is subjected to a significant plastic strain. Since plastic strain is known (6.48) to accelerate low temperature sensitization, the possibility exists that IHSI may have a somewhat detrimental effect on the susceptibility of treated welds over a long period of time from a metallurgical viewpoint. However, the chief concerns in assessing the effectiveness of IHSI are the possibility that in some locations and under some upset conditions the favorable residual stress state produced by IHSI may shake down and the possibility that preexisting cracks may be too deep for IHSI to produce significant benefits. Despite these possible drawbacks, it appears from the available analyses and experimental data that for most piping locations IHSI can significantly reduce susceptibility to IGSCC. The Task Group also recommends the use of IHSI on weldments joining a resistant piping material such as Type 316NG with pumps, valves, vessel nozzels, etc. of a nonresistant material.

Although both IHSI and LPHSW share the same fundamental basis as remedy procedures and experimental measurements (6.46, 6.49, 6.50) and analytical results (6.51) indicate that LPHSW and HSW are effective in producing compressive residual stresses on the inner surface of piping butt weldments, the existing data base of experimental results and finite-element analyses is much greater for IHSI than for LPHSW or HSW and more work is needed to demonstrate the applicability of LPHSW and HSW to other weldment geometries. In addition, both LPHSW and HSW appear to be more strongly dependent on weldment geometry and weld process parameters than IHSI. Thus it is difficult to consider LPHSW and HSW in a generic fashion. The use of HSW in making repairs and in fabrication of piping assemblies appears to be good engineering practice, and it has been widely used in Japan especially for partial pipe replacements (6.53), but because of the sensitivity to actual practice both LPHSW and HSW can be considered only partially effective remedies for IGSCC in conventional Type 304 stainless steel piping systems.

6.4.4 Evaluation of Water Chemistry Improvement

Since environment is one of the three synergistic contributors to IGSCC, the possibility of modifying the BWR water chemistry to prevent or reduce the chances of future occurrences of IGSCC has received considerable attention over the last several years (6.54). Modifications to the environment that have been proposed fall into two general classes, which should be considered both independently and in conjunction with one another: first, controls over the ingress of ionic species that can accelerate both initiation and propagation of cracks; and second, control of the electrochemical potential of the stainless steel by reducing the concentration of oxygen in the coolant through injection of hydrogen.

6.4.4.1 Current Regulatory Positions

In 1973 and 1974, the staff issued R.G. 1.56, "Maintenance of Water Purity in Boiling Water Reactors," and in 1977, issued Revision 1 to this Guide for comment. In each of these versions of the Guide, it was accepted that the oxygen concentration in BWRs was fixed by the radiolysis of water in the reactor core and the boiling process. Controls were placed on the ingress of impurities in the water to reduce the occurrence of SCC of sensitized stainless steel. The Guide specifies maximum conductivities of the coolant for long-term operation and gives absolute limits for short-term operation; if the latter are exceeded, the reactor must be shut down. These positions are generally consistent with the Technical Specification requirements for most operating BWRs. No mention is made in either version of the Guide on techniques for controlling oxygen.

6.4.4.2 Role of Impurities on Initiation and Propagation of IGSCC

Recent research at General Electric (6.55) and at Argonne National Laboratory (6.56) has shown that, within the water purity limitations spelled out in Regulatory Guide 1.56, there are significant effects of dissolved anionic impurities on both the initiation and the propagation of IGSCC. The three most common anions that can concentrate in films and produce electrolytes in the crack are chloride, which frequently enters a BWR coolant through condenser leakage, sulfate or sulfite, which can enter the coolant from condenser leakage or resin breakup, and carbonate, which can enter the coolant from air or water leakage into the condenser. The material itself may also provide a sufficient source of impurities, as noted in Section 2.

Further, crack-tip chemistry may differ significantly from bulk chemistry, through corrosion processes. Breakup of demineralizer resins can cause injection of the sulfonate cation resin beads, which decompose in the heat and radiation in the reactor to sulfites or sulfates, depending upon the oxidizing potential. If resin beds have been used for an extended period of time, particularly on a coastal plant, their breakup may inadvertently introduce chlorides that had been adsorbed in the resin beds, along with the sulfur species, to the coolant.

All three of these species have been shown to have significant accelerating effects on both initiation and propagation of IGSCC (6.54-6.56). Although tightening the allowable limits on these species (or on conductivity) in the BWR coolants could have a beneficial effect on the occurrence of IGSCC, it may not always be easy for a utility to operate with significantly tighter controls over a long period of time. EPRI has initiated a three-year program (6.57) which together with NRC-sponsored research (6.56) may provide a fundamental basis upon which improved water chemistry specifications can be established. It is known from laboratory tests that chloride ions, for example, once introduced into the coolant, will adsorb in the films near the growing cracks and are not easily removed following elimination of the chloride in-leakage into coolant. How much chloride, sulfate, and even carbonate are codeposited or adsorbed in the crud deposits is not known at the present time, nor is their role on subsequent cracking well understood. Savannah River (6.58), in a slightly different reactor some years ago, identified high concentrations of chloride in the crud deposits (which in those reactors were primarily hydrated aluminum oxides), despite maintenance of less than 0.2 ppm Cl^- in the reactor coolant. They did not establish, however, that Cl^- was a major contributor to the IGSCC that occurred in the Savannah River reactors. Consequently, it may not be possible to reduce significantly the occurrences of IGSCC solely by reducing the steady-state levels of chloride, sulfate, and carbonate in solution. Further, because the mechanisms of chemisorption and reactivation of the adsorbed species are not well understood, it is not possible to establish at this time whether a short period of high impurity levels in an otherwise impurity-free environment is more or less detrimental to a system than a relatively low but continuous source of impurities over a long period of time.

It is difficult to demonstrate a clear benefit from operation with low impurity levels from current plant operating data. Some plants using Powdex demineralizers which are known to operate with their water chemistry close to the allowable chloride or conductivity limits have fewer incidents of IGSCC than other units with "better water chemistry" (i.e., lower average conductivity levels) achieved using deep bed demineralizers (6.57, 6.59). Differences in materials and stress patterns may override differences in water chemistry between operating units. However, the laboratory data clearly indicate that for a given material and loading conditions, additions of sulfates, chlorides, or carbonates have a significant detrimental effect.

6.4.4.3 Role of Oxygen Control

Since IGSCC of sensitized stainless steel can occur in high purity water containing oxygen, chemical controls for prevention of IGSCC in a BWR require a reduction in the levels of both the ionic species and the oxygen in the primary coolant. If sufficiently low levels of oxygen are achieved, the electrochemical potential of the stainless steel will be shifted to a region at which IGSCC will neither initiate nor propagate. The electrochemical potentials required to prevent IGSCC depend strongly on the level of ionic impurities present (6.56). Current data suggest maintaining the electrochemical potential to less than -350 mV on the standard hydrogen scale and the conductivity of the coolant to less than 0.2 μS/cm (6.56). In Dresden-2, reducing the dissolved oxygen level to approximately 20 ppb brings the potential into this range. Large shifts in potential have been achieved by the injection of hydrogen into the BWR coolant (6.54, 6.59). Demonstrations of this technique have been made in Sweden at Oskarshamn and Ringhals and in the U.S. at Dresden-2 (6.54, 6.59-6.61) in which a corrosion test loop is deliberately tapped into one of the recirculation lines to monitor the corrosion potential of sensitized stainless steel and the behavior of stress-corrosion cracks in this environment.

In the hydrogen water chemistry approach, hydrogen is injected into the feedwater lines to reduce the residual oxygen level to approximately 20 ppb. When the coolant is pumped into the core of the reactor, this hydrogen suppresses the radiolytic production of oxygen in much the same way that hydrogen overpressure in a PWR prevents the accumulation of residual oxygen in that coolant. We have many plant-operating years of experience with a PWR primary coolant that demonstrate conclusively that reducing the oxygen and therefore the electrochemical potential of the stainless steel by this technique (together with strict controls on halide ions) eliminates the occurrence of IGSCC. However, the oxygen levels in PWR coolant systems are significantly lower in most cases (typically less than 5 ppb), than are practically achievable in U. S. BWRs (typically less than 20 ppb). Both in Sweden and at Dresden-2, short-term tests have shown that the SCC can be stopped: after seven months' operation at Dresden-2 with known UT indications in the piping, subsequent inspections showed no change in the ultrasonic signals from this IGSCC, but whether this can be attributed to the hydrogen water chemistry is not certain. Also, in the corrosion test loops, cracks already initiated stopped propagating when the hydrogen was injected (6.59). Long-term demonstration tests are planned under EPRI sponsorship at Dresden (and possibly at another BWR yet to be selected) to determine the overall effects on the plant system of hydrogen water chemistry (6.57). Already it is known that the same levels of hydrogen concentration to the feedwater have a different effect on the corrosion potential at Dresden than they do at Ringhals (6.54, 6.59). An argument has been made that these differences may be traced back to the fact that the Swedish BWRs do not have jet pumps (6.59). There are also possible differences due to other chemical impurities in the water, since a number of ionic species in the coolant are known to affect the ability of hydrogen at a given concentration to suppress production of radiolytic oxygen (6.62). Plant-specific water chemistry should therefore be strongly considered in the selection of a second plant for a demonstration test, and for selection of the hydrogen levels needed for mitigating IGSCC.

The principal known side effect of the hydrogen chemistry is that it produces increased levels of hydrogen and nitrogen-16 in the steam. While nitrogen-16 is present in all reactor primary coolants, from n,p reactions on the oxygen-16 in the water, the normal excess of oxygen in the coolant reacts with it to form an oxide of nitrogen or nitric acid, which is soluble and relatively nonvolatile. With the hydrogen chemistry, however, a reducing potential exists, which causes a portion of this nitrogen to be reduced to volatile ammonia gas, which enters the steam. Radiation levels in the turbine area at Dresden-2 have increased routinely by a factor of 5 when the hydrogen chemistry is operating. In fact, GE suggests that monitoring the radiation levels in the turbine provides a good indication of the effectiveness of the hydrogen treatment (6.59). For those plants with currently unshielded turbines, the introduction of hydrogen water chemistry may require turbine shielding. Access to the turbine (or unshielded areas) may require that the hydrogen be turned off. The stress-corrosion tests in the Dresden loop have shown that hydrogen interruptions lasting up to six hours do not appear to reinitiate IGSCC. However, during hydrogen interruptions of 18 hours or more, some IGSCC propagation can occur, but at a rate no greater than would occur in the absence of the hydrogen (6.59). Therefore, even if the hydrogen is operable only 90% of the time, the extent of crack propagation should be reduced by at least a factor of 10.

Other side effects of the hydrogen chemistry that need consideration are effects on the fuel cladding, reactor internals, and other plant components. Some years ago, PWRs switched from zircaloy-2- to zircaloy-4-clad fuel to minimize the hydrogen uptake in the fuel. Whether or not the hydrogen water chemistry approach would require a similar change with the zircaloy fuel cladding alloys in BWRs remains to be demonstrated. It is planned that fuel from Dresden-2, which will have been exposed to three full operating cycles under the hydrogen water chemistry, will be examined, under EPRI sponsorship in a cooperative program between EPRI and the utility, to determine the effects of hydrogen chemistry on the fuel cladding material (6.57). These effects are not anticipated to be large.

Reduction of the oxygen levels in the feedwater lines could lead to increased corrosion of the carbon steel components, which can accelerate the crud transport problem in BWRs. With the generally leaky condensers that exist in this country, there should probably be sufficient oxygen (20 ppb) in the feedwater to continue the passivation of the carbon steel. In Japan, where tight condensers appear to be the rule, they have had to inject oxygen into the demineralizer outlet to maintain sufficient levels for passivation of carbon steel even in the absence of hydrogen water chemistry. Increased amounts of hydrogen in the steam have also caused a potential for hydrogen fires, and oxygen injection into the steam lines to react with the hydrogen in the recombiners has been proposed.

6.4.4.4 Startup Deaeration

The probability of IGSCC may be greatest during the startup of a BWR when thermal stresses can cause creep of the pipes, relatively high oxygen levels are normally present, and the temperature is in the range of maximum susceptibility to IGSCC (150 - 250°C) (6.63, 6.64). Deaeration of the coolant during startup can be achieved by pumping on the hot well. While this alone cannot prevent IGSCC, startup deaeration in combination with the hydrogen chemistry should go a long way to minimize the occurrence of this problem in BWRs.

6.4.4.5 Foreign Experience (6.65)

The activities in Sweden towards development and testing of improved water chemistries have been discussed above. While most regulatory agencies set limits on impurity levels, similar to those in Regulatory Guide 1.56, actual operating levels of ionic species are generally lower in other countries than in the U. S. because of greater attention to condenser design and tighter controls on condenser leakage.

The Japanese believe their controls on water purity, combined with their emphasis on stress-improvement and use of alternative materials, are adequate, and they do not at present propose to use hydrogen injection to control oxygen in BWRs.

6.4.4.6 Role of Other Impurity Intrusions: Decontamination

From the above discussions as well as from those made earlier in Section 2, it is apparent that any action which disturbs chemically the protective layers of oxides on the surface of the stainless steel piping in the vicinity of sensitized stainless steels can have an effect on IGSCC. Although research has shown that relatively mild decontamination solutions, such as those used in the CANDECON process, do not increase the propagation rates of IGSCC in sensitized stainless steel in the laboratory (6.66), the effects of decontamination (removing the crud deposits and activating any ionic species that might be adsorbed in them) on initiation of SCC following return to service of the cleaned unit need to be much more thoroughly investigated. Many utilities faced with high radiation fields and frequent in-service inspections have felt it prudent to reduce personnel exposures by decontaminating the piping before the inspections. The long-term effects of this practice need to be investigated.

6.5 CONCLUSIONS AND RECOMMENDATIONS

The conclusions and recommendations from this chapter are grouped below under the major subheadings of the chapter.

6.5.1 Flaw Evaluation and Leak-Before-Break

Conclusions

The evaluation of flaws found in service and the validation of leak-before-break concept requires reasonably accurate knowledge of the ability to size the length and depth of SCC accurately, the applied and residual states of stress, the relation of crack growth rate to stress and environment, and the ultimate load-carrying capacity of the cracked pipe. A realistic evaluation of flaw evaluation procedures and the concept of leak-before-break requires an integrated approach considering all the above factors.

- The analysis of stress-corrosion crack growth in weldments using the techniques of linear elastic mechanics with "typical" values for the distribution of throughwall residual stresses and stress-corrosion crack growth rates yields results which are consistent with field and laboratory experience with welded pipes. Thus it is a reasonable approach to use for the analysis of the remaining life of flawed weldments in-reactor and for detailed probabilistic models such as the work by Lawrence Livermore National Laboratory.

- The combination of residual stress distributions and crack growth rates currently being used by NRR should yield predictions of flaw growth in weldments that are conservative in the great majority of cases. However, because of the wide variability in residual stress distributions near welds and IGSC crack growth rates, the current approach used by NRR is not considered unduly conservative for current BWR piping materials and environments. The Task Group endorses the guidelines for the evaluation of flaw growth in weldments given in draft NUREG-0313, Rev. 2.

- The Task Group also wishes to point out, however, that although conservative estimates of the throughwall growth of stress-corrosion cracks can be made with a high degree of confidence given an initial crack size, there is a high degree of uncertainty in the measurement of the depth of existing cracks, as discussed in Section 4, and this uncertainty must be adequately addressed in the development of acceptance criteria for flawed piping.

- The use of loading conditions defined by the ASME Code in performing flaw evaluations is acceptable, with the exception that SLA thermal expansion stresses should be included in the flaw evaluation when limit load is not achieved prior to significant crack extension. Use of these loading conditions is consistent with the approach which has been accepted by the NRC for the design and operation of other reactor components and systems.

- Analyses for large surface cracks, conducted using the relatively low ductile fracture resistance properties of stainless steel weld metal, show that the margins against fracture may not be as great as those believed to be associated with IWB-3640. Nonetheless, a substantial margin against fracture can be demonstrated for these flaws under normal operating conditions.

- Evaluation of throughwall cracks, using weld metal properties, shows that a surface crack breaking through the pipe wall under bounding accident loading conditions would have to be approximately 30% of the circumference in length, for large-diameter pipes, to result in unstable fracture of the pipe. However, leaks from cracks not large enough to cause gross failure of the pipe may be undesirably large.

- On the basis of fracture mechanics evaluation for bounding and typical stress conditions and weld toughness properties, it is concluded that IWB-3640 provides an adequate basis for evaluating the majority of the weld connections in BWR recirculation piping. This is especially true because many of the cracks will be in higher toughness zones adjacent to the lower toughness welds.

- Specific assessments of piping integrity will have to recognize weld conditions having lower toughness than the 304 stainless steel welds used in the analyses described in this report.

- Operating experience suggests that leak-before-break is the most likely mode of failure for the vast majority of cracks occurring in service. This is a result of the asymmetry of the weld residual stresses and applied loads and the variability in material properties. Evaluations using conservative crack growth rate predictions and net section collapse analyses, applicable to cracks in very high toughness material, indicate that for the vast majority of possible crack geometries there exists significant time to allow for detection of leakage and implementation of corrective actions. Evaluation using the fracture resistance properties of the weld material show substantial margins against failure, under normal and accident loading conditions for throughwall cracks which should be reliably detected by leakage.

Recommendations

Based on the above discussions and conclusions, the Task Group has developed the following recommendations.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair. The limitation on acceptable crack length is primarily a result of the lack of confidence in flaw depth sizing capability, and is intended to ensure leak-before-break conditions. The maximum allowable throughwall crack length can be determined based on weld joint specific loads.

- The maximum crack length allowable without repair for a specific weld joint should be the minimum of either 1) the throughwall crack length demonstrated by elastic-plastic fracture mechanics analyses to be stable under normal operating plus SSE loading conditions, 2) the throughwall crack length that would still permit the pipe to withstand normal operating plus SSE loading conditions as demonstrated by net section collapse (limit-load) analyses, or 3) the maximum crack length that would result in a leak rate greater than the plant's normal makeup capacity. Shorter cracks can be evaluated using the IWB-3640 criteria as modified by the NRC staff in SECY 83-267C. Calculations indicate that in the majority of cases the maximum crack length acceptable under the above criteria will be approximately 25% to 30% of the pipe circumference.

- The recommendation for a limit on acceptable crack length is primarily a result of the lack of confidence in ultrasonic depth sizing capabilities. In this respect, the fracture mechanics calculations presented in this report demonstrate that there are acceptable margins against fracture for relatively long, deep surface cracks. Demonstration of reliable sizing of the part-through flaw depth would most likely allow relaxation of the limits on crack length.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair.

- Additional fracture mechanics analyses, material properties characterization, and large-scale pipe tests should be performed to further our understanding of the implications of stainless steel weld and cast material fracture toughness properties in flawed pipe evaluations. Furthermore, the Task Group recommends active NRC support of the ASME task group currently evaluating the concerns which have been raised regarding IWB-3640.

- Since operating experience and fracture mechanics evaluations indicate that leak-before-break is the most likely mode of piping failure, it is recommended that reasonably achievable leak detection procedures be in effect in operating plants. Current sump pump monitoring systems are sensitive enough to provide additional margin against leak-before-break if more stringent requirements on surveillance intervals and unidentified leakage are imposed (see Section 4). Therefore, the Task Group recommends that the limits on unidentified leakage in BWRs be decreased to 3 gpm and that the surveillance interval be decreased to 4 hours or less.

6.5.2 Short-Term Solutions

- For relatively short axial cracks, analysis can be used to justify long-term operation with weld overlays, since errors in crack depth measurement or flaw growth predictions for these cracks will lead at worst to relatively small leaks, which will be easily detectable long before the crack can grow long enough to cause failure. For circumferential cracks, weld overlay is considered an acceptable repair procedure for a maximum of two refueling outages unless reliable techniques for the sizing of cracks through the overlay or for the monitoring of crack growth are developed.

6.5.3 Long-Term Solutions and Safety Assessment

- Experience with materials to mitigate IGSCC, such as 347NG in Germany and 304NG and 316NG in Japan, has been excellent. Other materials used in the U. S. include 304L and 316L. All of these alloys are more resistant to IGSCC than conventional Types 304 or 316 stainless steel. Based on U. S. data and prior use, Type 316NG stainless steel offers an additional margin of resistance to IGSCC and utilities that choose to replace pipe should be strongly encouraged to use it.

- The use of additional mitigating procedures such as IHSI or hydrogen water chemistry to provide additional margin against environmentally assisted cracking is strongly recommended even after replacement with Type 316NG stainless steel. Utilities should be encouraged to adopt an ALARA approach to coolant impurity levels.

- Although low-carbon stainless steels with nitrogen additions have been successfully fabricated and welded in Japan and Europe, U. S. experience with these materials is limited. It appears that greater care must be exercised in the control of composition and fabrication variables to limit cracking during hot forming or welding.

- IHSI is considered to be a more effective mitigating action for IGSCC than HSW and LPHSW in part because more data are available to demonstrate that the process does produce a more favorable residual stress state. All the residual stress improvement remedies are considered to be more effective when applied to weldments with no reported cracking.

- IHSI is an acceptable mitigating procedure for weldments with no reported cracking even in older plants. However, to be fully effective a suitable hydrogen water chemistry also should be implemented.

- The use of IHSI on weldments with detectable cracking must be considered on a case-by-case basis. However, for relatively short cracks (approximately 20% of the circumference in length), since even large errors in crack sizing or the prediction of flaw growth will lead only to small leakage, the decision on whether an additional repair is required can be determined by analysis. For longer cracks, repair will probably be required.

- All steps related to the weld prep, welding, and weld finishing and HSW or IHSI should be validated through certification and through use of an appropriate QA manual. As noted previously, the weld should be optimized for future ISI.

- BWR water chemistry controls should be modified to minimize IGSCC. These modifications should include both a substantial reduction in the levels of ionic species entering the primary coolant and a control of oxygen level. The current work on reduction of oxygen through hydrogen additions should be followed closely with the possibility that it may be employed to reduce further the electrochemical potential of the stainless steel to a level at which SCC, either IGSCC or TGSCC, will not occur. It appears that hydrogen water chemistry is an effective IGSCC countermeasure. However, ongoing work regarding potential adverse effects on other reactor components should be closely followed in order to confirm the acceptability of this countermeasure.

- Either reduction of ionic species or hydrogen water chemistry, used alone, will have some beneficial effects. Used in combination, they show great promise for mitigating IGSCC.

- The effects of hydrogen water chemistry on the balance of the system and on overall long-term plant operation need to be explored in greater depth before they can be recommended without reservation for adoption on a wide scale.

- Regulatory Guide 1.56 needs to be modified to reflect these water chemistry recommendations to provide tighter controls on water conductivity and provisions for oxygen control as an acceptable option.

- The long-term effects of decontamination procedures on IGSCC need to be investigated.

In the event that any of the preceding recommendations of this section are implemented, there may be a need to modify the following documents when relevant to the specific issue: NUREG-0313 Revision 2; Regulatory Guides 1.45 and 1.56; Standard Review Plans 5.23 and 5.4.8, Plant Technical Specifications Governing Unidentified Leakage Rate Limits and Leakage Surveillance Requirements; 10 CFR Appendix A, Nos. 31, 32; Regulatory Guides 1.31, 1.44, 1.116; SRP 3.6.1, 3.6.6, 5.2.3; ASME XI.

6.6 SUMMARY

Section 6 represents a complex interaction of cracking analyses, decisions on short-term remedies and longer-term permanent remedies. Table 6.3 is an attempt to collect the available options into one table, covering the various conditions of plants not yet built, plants ready for licensing and plants operating for several years, typical of these now experiencing IGSCC where the options include no evidence of cracking cracks believed to be shallow in the sense of ASME XI and deep cracks. The purpose of Table 6.3 is to permit the various options to be examined concurrently.

TABLE 6.3

Recommended Measures for Controlling IGSCC in BWR Piping

	Near Term			Long Term		Ultrasonic Exam.			
BWR Plant Status	Overlay Weld*	Replace with Similar Alloy	Residual Stress Improvement IHSI, HSW	Hydrogen Water chemistry	316 NG Piping.	New Baseline	Accel. UT	Norm. UT	Enhanced Leak Detection (moisture tapes, AE, etc.)
Design Stage			X	X	X			X	
NTOL or Recent Startup			X	X		X		X	
Operating >5(?) years									
No Cracks Detected			X	X		X	X------X#		X (maybe)
Only Limited Shallow Cracks			X	X		X	X*------X		X
Deeper Cracks -- Few or Many	X	X maybe	X**	X	X	X	X then	X (for 316NG)	X Prior to Replacement

*Limited to two cycles unless convincing evidence is presented

**Also suggested after replacement with 316 NG

Prior to mitigation, accelerated UT; thereafter normal UT

X Accelerated UT limited to cracked welds

* With mitigation, accelerated UT limited to cracked welds

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7.0 IGSCC RISK STUDIES

7.1 INTRODUCTION

Earlier studies on IGSCC in BWR piping systems approached the impacts primarily on a deterministic basis which defines a spectrum of design basis accidents (DBA) considered credible. The nuclear power plant must be capable of accommodating these accidents within the envelope of acceptable consequences to the public. With sufficiently reliable engineered safety features, the risk of an unmitigated accident will be an acceptably small contribution to the residual risk, which lies outside the envelope defined by the design basis. Probabilistic Risk Assessments (PRA) and risk-related methodologies are being increasingly utilized to quantify the residual risk to the public. PRA provides a systematic method to examine how this residual risk is structured and for identifying the important contributors.

Several plant PRAs have been completed since the pioneer Reactor Safety Study (RSS) in late 1975. Risk methodologies have been utilized in a number of useful ways in the regulatory process. These risk studies are intended to evaluate the LOCA risk impact of IGSCC as a disciplined approach to relate field and inservice inspection experience to risk impact, and as one input to the value-impact studies of Section 8.

A rigorous understanding of the impacts of IGSCC on LOCA frequency is not known. IGSCC is only one cause of pipe failure and the proportionality between these causes can only be inferred from field experience at this time. Lawrence Livermore National Laboratory (LLNL) is currently conducting a study for the NRC which should improve our understanding of how IGSCC affects pipe failure frequency, including its relationship to other causes. In the absence of such an understanding, it was decided to initiate these studies by testing the sensitivity of representative plant PRAs to the LOCA contribution. This led to a confirmation of similar studies performed by the Electric Power Research Institute (EPRI) as well as NRC in-house investigations. With the understanding of the sensitivity of risk to LOCA contribution, an attempt was made to assess whether the evidence - analytical and/or statistical (field data) - implies an IGSCC contribution to risk that is not accounted for in current analyses.

7.2 LOSS-OF-COOLANT-ACCIDENT (LOCA) RISK

The risk methodology for LOCA calculations was defined in the RSS and has remained relatively unchanged in subsequent PRAs. Risk is defined as the product of the probability of an event and the consequences of the event. It has become common practice in sensitivity studies to assume that risk is proportional to the core damage frequency, rather than to trace the radioactivity release pathways for each differential change to the core damage frequency. This approach has been taken in this study.

In Appendix D.2, a simple sensitivity model based on the RSS methods is developed. Review of pertinent BWR PRAs, as shown in Table 7.1, demonstrates that LOCA contributions to the core damage frequency are not considered

Table 7.1

Sensitivity of Core Melt Frequency for BWR PRAs
for a Postulated Increase (10x) in LOCA Initiator Frequency

Plant A*	Sponsor	Total Core Melt Frequency** (per Rx yr)	Frequency of Core Melt Due to LOCA Initiated Sequence (per Rx yr)	Increase in the Frequency of Core Melt Due to Postulated Increase (10x) in LOCA Initiator Frequency
Limerick (Rev 5)	PE Co.	1.5 E-5	1.3 E-7	1.2 E-6
Shoreham	LILCO	5.5 E-5***	1.9 E-6	1.7 E-5
Millstone 1	IREP	3.0 E-4	3.0 E-6	2.7 E-5
Browns Ferry	IREP	2.0 E-4	1.2 E-6	1.2 E-5
Peach Bottom (RSS)	NRC	2.9 E-5	1.2 E-6	1.1 E-5

*Big Rock Point was considered but deemed atypical, since it is a BWR-1 and has a less comprehensive ECCS. GESSAR (Grand Gulf) was also considered, but was judged inappropriate for this list because it is a BWR-6, among other reasons.

**Total core melt frequencies reported in the probabilistic analyses used for this study resulted from internal initiators only. Analyses including external initiators would yield higher core melt frequencies and, thus, a lower percentage contribution from LOCAs.

***The Shoreham value for "core vulnerable frequency" is believed to be representative of core melt frequency and has been used here.

dominant contributors. The five plants exhibit an average of 2% LOCA contribution to the total core melt frequency. A sensitivity analysis shown in Table 7.1, using the sensitivity model developed in Appendix D, indicates that even an order of magnitude underestimation in the LOCA frequency would not cause the LOCA risk to be a dominant contributor to core damage frequency. For example, for Millstone 1, the LOCA contribution would change from a 1% core melt frequency contribution to a 10% contribution.

The field data impact on the frequencies of the three traditional BWR LOCA sizes was examined to see if the precursor IGSCC has resulted in a statistically significant error in the LOCA frequencies used in PRAs and thereby cause underpredicted risk contributions. To date there have been no large primary recirculation pipe failures in the U.S. commercial nuclear power industry. The implication of a significant underprediction of the large LOCA frequencies is examined in Appendix D.3. It is found that a significant (factor of 10) increase in the large LOCA frequency, say, due to IGSCC or other causes, would have been evidenced by several failures for the 273 reactor years of U. S. BWR operation. Currently, 700+ reactor years of operation in the U. S. commercial nuclear power program have resulted in no large pipe failures in the primary coolant system. The likelihood of an unobserved underprediction of large LOCA frequencies is even further reduced if the foreign BWR experience is utilized. The aggressive inservice inspection and repair/replacement program obviously modifies the empirical failure frequencies to even lower values, but the extent of this effect is not clear.

In the case of intermediate and small LOCAs, the data base is examined by a review of both current studies and the LER system to determine if statistically significant time trends would indicate an unanticipated IGSCC contribution to these frequencies. Although the quality of the data base as given by the LER system is considerably below experimental standards, no significant time trend increase could be attributed to IGSCC. From these studies it is concluded that it is highly unlikely that the appraisal of the public risk is significantly (an order of magnitude) in error.

7.3 SYSTEMS RISK EVALUATIONS

The risk implications on a systems basis is considered in Appendix D.4. Systems contribute in two ways to PRA considerations. Those portions of systems which are nonisolable from the primary coolant system comprise part of the LOCA sensitive piping, and those systems which mitigate accidents (engineered safety features) directly contribute to the reduction of risk. Risk methods can now rank the importance of systems according to their contribution to reducing risk, such as shown in Table 7.2. Risk achievement ratio is defined as the factor by which core melt frequency would increase if the system were never operable. This table indicates the importance of the Residual Heat Removal System (and thus the integrity of its piping) in risk reduction. Long-term heat removal is required in all sequences.

Table 7.2

Risk Worth for Safety Systems* With Respect to Core Melt Frequency

System	Risk Achievement Ratio
Reactor Protection System	2.0×10^5
Residual Heat Removal System (RHR)	1500
Standby Service Water System	1500
Low Pressure Coolant Injection (LPCI) System	23
Automatic Depressurization System (ADS)	14
Low Pressure Core Spray (LPCS) System	7.6
High Pressure Core Spray (HPCS) System	4.4
Reactor Core Isolation Cooling (RCIC) System	2.2

*Based on Grand Gulf, a BWR-6, which has HPCS rather than HPCI as in BWR-4 as well as some minor differences, but safety systems/functions are similar. Some systems not directly pertinent to LOCA concerns have been omitted from this list.

7.4 CONCLUSIONS AND RECOMMENDATIONS

7.4.1. Conclusions

- Intergranular stress-corrosion cracking (IGSCC) has caused no significant loss-of-coolant events (LOCE) or loss-of-coolant accidents (LOCA) in BWRs even though its presence in Type 304 stainless steel coolant piping has been found in many operating plants. This may be due to an aggressive inservice inspection and repair program and/or a tendency to leak-before-break (with effective leak detection). The LOCA contribution to the core damage frequency as calculated by probabilistic risk assessment (PRA) studies for BWRs is a minor contribution (generally less than 10%) for all LOCA sizes. Because the LOCA contribution to the core damage frequency is small, it would require a considerable error, at least an order of magnitude, from IGSCC before LOCAs would be a dominant contributor to the core damage frequency.

- IGSCC is but one component of the LOCA occurrence rate. Design and construction errors, maintenance errors, cyclic and thermal fatigue, etc. all contribute to the LOCA probability. It is estimated that IGSCC may be 30% or more of the LOCA frequency in BWRs as deduced from experience, the total industrial data base, and engineering judgment. Therefore, an error in the IGSCC LOCA contribution of 20-30 would be required for an order of magnitude error in the LOCA probability.

- The pipe failure data base and available studies were reviewed for evidence of IGSCC impact. To a considerable extent, uncertainties have always existed in determining piping failure rates. Largely, this uncertainty is intrinsic in attempts to measure low-frequency events from experience, and it makes the effect of trends difficult to analyze. It was statistically reasoned that an error of an order of magnitude in the large pipe failure frequencies (presumably from IGSCC) would have produced, with one chance in four, at least one failure during this interval. Therefore, it was concluded that the frequency for large pipe failure adequately accounts for the contribution due to IGSCC.

- For intermediate and small pipe failures, the incidence of reported and identified corrosion events in the data base, although increasing somewhat over time (owing to improved reporting and other causes), was judged to be not statistically indicative of an error in the failure frequencies.

- The systems aspects of IGSCC risk were revised using Hatch-2 as a model. It was concluded that the plants should be required to be designed for the most severe large LOCA without significant core damage. The ECCS system must be able to survive the most severe single failure in the system and still perform its intended function(s). Pipe failure is no more severe than other failures in the system (valves, pumps, etc.) and is probably less severe. Coupling this with the ECCS redundancy and conservative margin in both design and calculational procedures, it is concluded that IGSCC would cause little increase in risk from the standpoint of systems important to safety.

- Multiple failures were briefly considered. IGSCC may be a common mode failure in the recirculation piping and other systems. It is concluded that a large LOCA and ECCS system failure both caused by IGSCC is adequately protected by the single-failure criteria and systems redundancy. Multiple failures causing a LOCA larger than a DBA are likely to add little to the risk because it appears that multiple failures are several orders of magnitude less in probability than a large single failure.

- Overall it is concluded that the presence of IGSCC in BWRs may reduce safety margins believed to exist, but there is no apparent evidence to conclude that IGSCC has increased public risk significantly.

7.4.2 Recommendations

Piping failure rates are difficult to determine because failures of high quality piping occur infrequently and require a data base acquired over a large number of reactor years, since normal experimental procedures are impractical. This difficulty is unlikely to be overcome now or in the foreseeable future. What results is an estimated frequency with large uncertainties from aggregated events acquired from poor-quality documentation. To partition this frequency according to contributing phenomena in a meaningful way is very difficult.

- It is recommended that a formal study and updating of the pipe failure data base be undertaken. Some statistics may be only meaningfully derived or assessed analytically. The NRC has developed a computer code PRAISE (P iping R eliability A nalysis I ncluding S eismic E vents) which is currently being modified to include a model for IGSCC based on current understanding.

- It is recommended that PRAISE be used to a) investigate the impact of IGSCC in primary piping according to pipe size, b) study the failure frequencies of components due to IGSCC, fatigue, etc., and c) calculate the conditional probability of multiple failures associated with IGSCC to check the point value and to test the assumption of a lognormal distribution.

8.0 INTERGRANULAR STRESS CORROSION CRACKING (IGSCC) VALUE-IMPACT ASSESSMENT

8.1 INTRODUCTION

This report presents a value-impact assessment of alternative plant modifications and operating procedures to prevent and mitigate the effects of IGSCC in operating BWRs. The analysis was performed by Battelle Memorial Institute-Pacific Northwest Laboratories (PNL) and used methods developed in A Handbook for Value-Impact Assessment (8.1). The format and additional details on the development of the methods are discussed in the reference and not repeated here.

This assessment performs analysis on postulated generic scenarios for preventing or mitigating IGSCC in a typical BWR plant based on licensee actions and plans to date. Basically, three scenarios each with options involving use of Hydrogen Water Chemistry (HWC) have been selected for analysis. Two of the scenarios--pipe replacement and IHSI of uncracked pipe--are considered as potential long-term "fixes." The third scenario--augmented UT inspection and weld overlay repair--is considered only a short- or intermediate-term control. In general, the mitigation scenarios are graduated to the extent that the greater the assurance the scheme will prevent future IGSCC, the less stringent are the assumed post-mitigation inspection scope and frequency requirements. It is recognized that many plants may, in fact, adopt a long term mitigation plan which employs a combination of IHSI of uncracked pipe and replacing cracked pipe. It is anticipated that the actual scenarios will fall within the range of those considered here or some combination of them.

The basis of the impact assessment is utility, EPRI and vendor estimates of incurred costs, Occupational Radiation Exposure (ORE), and incremental plant downtime for implementation of inspection and repair activities performed to comply with IE Bulletins 82-03 and 83-02 plus similar estimates of utilities planning to replace IGSCC susceptible pipe.

The summary results and the conclusions from the value impact analysis are presented below. Appendix E presents a description of the scenarios and associated assumptions and discusses the analysis of each factor analyzed.

8.2 SUMMARY OF PROBLEM AND PROPOSED SOLUTION:

The results of the value-impact assessment of the postulated scenarios are in Tables 8.1 to 8.4 below. The attributes considered in the analysis are shown in the following chart:

AFFECTED ATTRIBUTES

Decision Factors	Causes Quantified Change	Causes Unquantified* Change	No Chang
Public Health	X		
Occupational Exposure (Accidental)	X		
Occupational Exposure (Routine)	X		
Public Property	X		
Regulatory Efficiency	X		X
Improvements in Knowledge			X
Industry Implementation Cost	X		
Industry Operation Cost	X		
NRC Development Cost	X		
NRC Implementation Cost	X		
NRC Operation Cost	X		

*In this context, "unquantified" means not readily estimated in dollars.

Table 8.1

Value-Impact Summary
Pipe Replacement

Summary of Problem and Proposed Solution:

Pipe replacement is proposed to eliminate IGSCC in a BWR. Estimates are listed without hydrogen water chemistry. The HWC option has minimal impact on this scenario and these estimates are tabulated in the conclusions. A discount rate of 5% was used.

	Dose Reduction (person-rem)			Evaluation ($) x 10^{6}[a]		
Attribute	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Estimate						
Public Health	151	605	76			
Occupational Exp.						
Accidental	NQ	NQ	NQ			
Routine	-1810	2225	-1520			
Offsite Property				NQ	NQ	NQ
Onsite Property				NQ	NQ	NQ
Reg. Efficiency				NA	NA	NA
Imp. in Knowledge				NA	NA	NA
Industry Cost						
Implementation				-44	-57	-35
Power Replacment (Implementation)				-41	-68	-27
Operation				-0.12	-0.23	-0.088
NRC						
Development				NQ	NQ	NQ
Implementation				NQ	NQ	NQ
Operation				NQ	NQ	NQ
TOTAL	-1660	-1620	-1445	-85	-125	-62

RATIO: Public dose Reduction/Sum of All NRC and Industry Costs

(person-rem/10^6)[b] = 1.8

NA = Not affected.
NQ = Not quantified.
(a) Note: Favorable or beneficial consequences of a proposed action have a postive sign. Unfavorable or adverse consequences have a negative sign. For instance, an increase in industry or NRC operating costs would be considered an unfavorable consequence and should be entered in the table with a negative sign.
(b) Strictly speaking, because the ratio would be expressed as a positive number, the analyst should use the absolute value of the sum of all costs (industry, NRC, and other) in the denominator.

Table 8.2

Value-Impact Summary
IHSI

Summary of Problem and Proposed Solution:

IHSI and hydrogen water chemistry are proposed to eliminate IGSCC in a BWR. The estimates listed are with water chemistry using a 5% discount rate. The estimates for IHSI without HWC are not significantly different and are tabulated in the conclusions.

	Dose Reduction (person-rem)			Evaluation ($) x 10^{6}(a)		
Attribute	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estima
Estimate						
Public Health	151	605	76			
Occupational Exp.						
Accidental	NQ	NQ	NQ			
Routine	-690	-1060	-500			
Offsite Property				NQ	NQ	NQ
Onsite Property				NQ	NQ	NQ
Reg. Efficiency				NA	NA	NA
Imp. in Knowledge				NA	NA	NA
Industry Cost						
Implementation				-9.0	-13.9	-4.0
Power Replacment (Implementation)				-23	-32	0
Operation				-4.4	-1.7	-7.3
NRC						
Development				NQ	NQ	NQ
Implementation				NQ	NQ	NQ
Operation				NQ	NQ	NQ
TOTAL	-539	-445	-424	-36.4	-47.6	-11.3

RATIO: Public dose Reduction/Sum of All NRC and Industry Costs

(person-rem/$$10^{6}$)(b) = 4.2

NA = Not affected.
NQ = Not quantified.
(a) Note: Favorable or beneficial consequences of a proposed action have a postive sign. Unfavorable or adverse consequences have a negative sign. For instance, an increase in industry or NRC operating costs would be considered an unfavorable consequence and should be entered in the table with a negative sign.
(b) Strictly speaking, because the ratio would be expressed as a positive number, the analyst should use the absolute value of the sum of all costs (industry, NRC, and other) in the denominator.

Table 8.3

Value-Impact Summary
Augmented Inspection and Repair with HWC

Summary of Problem and Proposed Solution:

Augmented inspection, weld repair, and the use of HWC are considered a partial intermediate solution of IGSCC in a BWR. The ranges listed are for 20%-40% inspection each outage. Money was discounted at 5%.

	Dose Reduction (person-rem)			Evaluation ($) x 10^{6}(a)		
Attribute	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Estimate						
Public Health	151	605	76			
Occupational Exp.						
Accidental	NQ	NQ	NQ			
Routine	-1094	-4504	-249			
Offsite Property				NQ	NQ	NQ
Onsite Property				NQ	NQ	NQ
Reg. Efficiency				NA	NA	NA
Imp. in Knowledge				NA	NA	NA
Industry Cost						
Implementation				-2	-3	-1
Power Replacment (Implementation)				0	0	0
Operation				-60	-87	-36
NRC						
Development				NQ	NQ	NQ
Implementation				NQ	NQ	NQ
Operation				NQ	NQ	NQ
TOTAL	-1059	-4575	-194	-62	-90	-37

RATIO: Public dose Reduction/Sum of All NRC and Industry Costs

(person-rem/$\$10^{6}$)(b) = 2.4

NA = Not affected.
NQ = Not quantified.
(a) Note: Favorable or beneficial consequences of a proposed action have a postive sign. Unfavorable or adverse consequences have a negative sign. For instance, an increase in industry or NRC operating costs would be considered an unfavorable consequence and should be entered in the table with a negative sign.
(b) Strictly speaking, because the ratio would be expressed as a positive number, the analyst should use the absolute value of the sum of all costs (industry, NRC, and other) in the denominator.

Table 8.4

Value-Impact Summary
Augmented Inspection and Repair - No HWC

Summary of Problem and Proposed Solution:

Augmented inspection, weld repair without HWC is considered only short-term control of IGSCC in a BWR. The ranges listed are for 40%-60% inspection each outage. Money was discounted at 5%.

Attribute	Dose Reduction (person-rem) Best Estimate	High Estimate	Low Estimate	Evaluation ($) x 10 Best Estimate	High Estimate	Low Estimate
Estimate						
Public Health	151	605	76			
Occupational Exp.						
Accidental	NQ	NQ	NQ			
Routine	-2255	-7747	-611			
Offsite Property				NQ	NQ	NQ
Onsite Property				NQ	NQ	NQ
Reg. Efficiency				NA	NA	NA
Imp. in Knowledge				NA	NA	NA
Industry Cost						
Implementation				0	0	0
Power Replacment (Implementation)				0	0	0
Operation				-125	-167	-85
NRC						
Development				NQ	NQ	NQ
Implementation				NQ	NQ	NQ
Operation				NQ	NQ	NQ
TOTAL	-2250	-8095	-535	-125	-167	-85

RATIO: Public dose Reduction/Sum of All NRC and Industry Costs

$(\text{person-rem}/\$10^6)^{(b)} = 1.2$

NA = Not affected.
NQ = Not quantified.
(a) Note: Favorable or beneficial consequences of a proposed action have a postive sign. Unfavorable or adverse consequences have a negative sign. For instance, an increase in industry or NRC operating costs would be considered an unfavorable consequence and should be entered in the table with a negative sign.
(b) Strictly speaking, because the ratio would be expressed as a positive number, the analyst should use the absolute value of the sum of all costs (industry, NRC, and other) in the denominator.

8.3 VALUE-IMPACT ASSESSMENT CONCLUSIONS

Cost and ORE results for the three scenarios are summarized below and in the summary sheets. Public safety impacts are the same for all cases and not repeated here.

	ORE (man-rem)			Present Value Cost ($M) 5%			Present Value Cost ($M) 10%		
	B	H	L	B	H	L	B	H	L
PIPE REPLC.									
W HWC	1830	2245	1540	91	130	70	90	129	68
W/O HWC	1810	2225	1520	85	125	62	85	125	62
IHSI									
W HWC	690	1060	500	36	48	11	35	47	9
W/O HWC	738	1198	498	37	56	3.3	35	52	3.2
AUGMENTED INSPECTION/REPAIR									
W HWC	1094	4504	249	62	90	37	42	61	25
W/O HWC	2255	7747	611	125	167	85	82	109	56

The results support the following conclusions:

- The analysis indicates that this issue has low public risk impact, based on risk sensitivity studies and a review of pipe failure data including statistical estimates of IGSCC contribution to failure.

- Each of the scenarios has low value-impact ratios. However, when considering 30-35 affected BWR plants, the total potential public risk reduction indicates a low to at most medium safety priority ranking using the guidance in NUREG-0933, "A Prioritization of Generic Safety Issues" (8.2).

- However, economics dictate taking some action to reduce IGSCC impacts. IHSI with or without hydrogen water chemistry (HWC), HWC alone, or pipe replacement are the order of cost preference. All options will reduce cost below the inspect and repair strategy alone. The results are not sensitive to discount rates except in the case of inspect and repair costs.

- HWC is not cost effective with pipe replacement and is marginal with IHSI. HWC in plants with existing cracking problems that prevent the use of IHSI is advantageous.

- Occupational Radiation Exposure (ORE) commitments for all strategies are significant. IHSI will minimize the impact. Inspection and repair, without HWC, will exceed pipe replacement in man-rem.

8.4 REFERENCES

8.1 S. W. Heaberlin et al. 1983. A Handbook for Value-Impact Assessments. NUREG/CR-3568.

8.2 R. Emrit et al. 1983. A Prioritization of Generic Safety Issues. NUREG-0933.

9.0 SUMMARY

CONCLUSIONS AND RECOMMENDATIONS

The current PCTG has reconsidered the subject of pipe cracking in nuclear power facilities and has reached conclusions and recommendations consistent with those reported in NUREG-0531 and NUREG-0313, Revs. 1 and 2. The conclusions and recommendations that follow present new ideas or amplify areas addressed in the above-listed documents.

The PCTG believes that the following are the most important results of this study:

2.0 CAUSES AND DESCRIPTION OF IGSCC PHENOMENA

Conclusions

- IGSCC in BWR piping is occurring owing to a combination of material, environment, and stress factors, each of which can affect both the initiation of a stress-corrosion crack and the rate of its subsequent propagation. In evaluating long-term solutions to the problem, one needs to consider the effects of each of the proposed remedial actions in terms of the current understanding of the initiation and propagation of stress-corrosion cracks.

Recommendations

- Mitigating actions to control IGSCC in BWR piping must be designed to alleviate one or more of the three synergistic factors: sensitized material, the convention BWR environment, and high tensile stresses.

- Because mitigating actions addressing each one of these factors may not be fully effective under all anticipated operating conditions, mitigating actions should address two and preferably all three of the causative factors; e.g., material plus some control of water chemistry, or stress reversal plus controlled water chemistry.

3.0 PRESENT STATUS OF PIPE CRACKING IN OPERATING BWRs

Conclusions

- Although significant IGSCC was found in the United States in the past, no cracking was reported in large pipes until recently. As a result of extensive inspections performed during the past two years, IGSCC has now been reported in 19 out of 23 plants. All other countries have experienced IGSCC in piping made of Type 304 stainless steel, including large-diameter Type 304 stainless steel piping, in those plants that have it.

- In all countries, the preferred long-term solution is to replace piping made of Type 304 stainless steel. Type 316NG is considered the preferred choice, except in Germany, where they prefer their special nuclear grade of Type 347.

- Interim or short-term fixes in other countries follow the same approaches as have been used in the United States, and include IHSI, weld overlay reinforcement, welded "clam shell" reinforcement, last pass heat sink welding, and interim operation with unrepaired small cracks. It should be noted that IHSI on uncracked Type 304 stainless steel pipe is considered to be a permanent fix in Japan.

- Hydrogen water chemistry is being seriously considered by several countries as a desirable adjunct to other solutions.

4.0 NONDESTRUCTIVE EVALUATION OF PIPING WELDS

Conclusions and Recommendations

(There are) significant problem areas relative to the uncertainty of UT inspection during both preservice and in-service examination. The following items represent a synthesis of important conclusions and recommendations:

- Code minimum UT procedures result in totally inadequate IGSCC detection. Easily implementable modifications to these procedures can result in some improvement. These have been incorporated into Code Case N-335. Therefore, it is recommended that Code Case N-335 should immediately be made mandatory for all augmented inspections until better procedures are developed.

- Although IGSCC detection has improved to the point that it is considered acceptable under optimum conditions and procedures, the detection reliability as impacted by variability in operator procedure and equipment performance along with field conditions needs further study and improvement. While length sizing of cracks is acceptable, depth sizing is currently inadequate. It is recommended that advanced techniques and procedures for crack detection and depth sizing continue to be developed and incorporated into Code requirements to provide data to reduce the need for extremely conservative fracture mechanics evaluation.

- The current activities in personnel and procedure qualification and performance demonstration represent steps in the right direction, and the resultant process that is being implemented is acceptable in the interim; however, they need further improvement. Therefore, it is recommended that ongoing industry and NRC activities to develop adequate criteria for qualification of the entire inspection process to achieve more reliable field inspection be completed and implemented on a high priority basis.

- For future plants or for replacement of existing piping systems, the material, design of pipe joints, and accessibility from both sides of the weld should be optimized for UT examinations; this requirement should be mandatory for all components with the exception of existing items such as pumps, valves and vessels in older plants. The uninspectable joints should be subjected to IHSI.

- Inspection techniques should be developed for detection and dimensioning of flaws in pipes repaired by the weld overlay process.

- Improved leak detection systems would permit more stringent requirements on unidentified leakage without increasing the occurrence of spurious shutdowns due to relatively benign leakage, and their development should be pursued.

- Acoustic and moisture-sensitive tape leak detection systems for local leak detection are available, and their use is recommended where inspection is difficult or impossible or operation for more than one fuel cycle is considered for long cracks with weld overlay repairs.

- An effort is also needed to be made to accumulate information on current acoustic leak detection (ALD) and moisture sensitive tape (MST) field installations. Detailed reports on the operability, maintenance, and reliability of these systems should be acquired, assembled, and distributed for review by utility, government, and research personnel interested in improving reactor leak detection technology.

- Sump pump monitoring systems have sufficient sensitivity to detect leak rates as small as 1 gpm in one hour. Improved leak detection systems would provide additional margin against leak-before-break without increasing the number of false alarms by providing information about leak location and leak source discrimination. It is, therefore, recommended that improved leak detection systems under development be completed and field tested.

- Since the major purpose of the preservice investigation (PSI) is to provide reliable NDE baseline for comparison with subsequent ISI, it is recommended that the latest edition of Section XI acceptable under 10 CFR 50.55(a) be used for the PSI.

- Expand modeling work to predict UT crack response for guiding development of UT techniques and guiding qualification test sample selection.

- Since the UT examiner is one of the more erratic inspection variables, it is recommended that human factors research be performed to lead to a reduction in the human variability.

5.0 INSPECTION OF PIPING FOR IGSCC

Conclusions

- Augmented inspections beyond those required by ASME Code Section XI should apply to all Types 304 and 316 austenitic piping systems operating over 200°F (93°C), unless they have been treated with effecting countermeasures.

- The degree of augmented inspection is dependent on the material and processes used for each weld. The most frequent inspections are required for welds fabricated from nonresistant material.

- Field data show that the cracking experience does not correlate well with the stress rule index and the carbon content. Therefore, the primary bases for sample selection should be field experience coupled with other factors such as weld prep, excessive grinding, extensive repairs, or high-stress locations.

Recommendations

- All Types 304 and 316 austenitic piping systems operating over 200°F (93°C) should receive augmented inservice inspection, unless they have been treated with effective countermeasures.

- The extent and frequency of examinations should depend on the resistance of materials to IGSCC and the effectiveness of any processes used to prevent cracking.

- The primary basis for sample welds selected for examination should be field experience not the stress rule index and carbon content. Other factors such as weld prep, excessive grinding, extensive repairs, or high-stress locations should also be considered in the sample selection.

- The performance capability demonstration program used to demonstrate the effectiveness of UT equipment, procedure, and examiner combination on the service-induced cracked samples should be continued and strengthened. The effectiveness and reliability of detecting IGSCC with all future UT procedure/examiner combinations should be demonstrated on the cracked samples before field application.

- All UT examiners should attend industry-sponsored UT detection and sizing training courses and should continue to be evaluated on the cracked samples under the witness of a third party personnel before participating in the field inspection.

- All BWR weld examinations should be performed in accordance with the latest version of Code Case N-335 and with the specific equipment and procedures used and personnel passed in the performance demonstration tests.

- An appropriate local monitoring system should be used to monitor the continuous integrity of the uninspectable welds in categories B and C. These local monitoring systems should be approved by NRC before field applications.

6.0 DECISION AND CRITERIA FOR REPLACEMENT, REPAIR, OR CONTINUED OPERATION WITHOUT REPAIR

6.1 Flaw Evaluation and Leak-Before-Break

Conclusions

The evaluation of flaws found in service and the validation of leak-before-break concept requires reasonably accurate knowledge of the ability to size the length and depth of SCC accurately, the applied and residual states of stress, the relation of crack growth rate to stress and environment, and the ultimate load-carrying capacity of the cracked pipe. A realistic evaluation of flaw evaluation procedures and the concept of leak-before-break requires an integrated approach considering all the above factors.

- The analysis of stress-corrosion crack growth in weldments using the techniques of linear elastic mechanics with "typical" values for the distribution of throughwall residual stresses and stress-corrosion crack growth rates yields results which are consistent with field and laboratory experience with welded pipes. Thus it is a reasonable approach to use for the analysis of the remaining life of flawed weldments in-reactor and for detailed probabilistic models such as the work by Lawrence Livermore National Laboratory.

- The combination of residual stress distributions and crack growth rates currently being used by NRR should yield predictions of flaw growth in weldments that are conservative in the great majority of cases. However, because of the wide variability in residual stress distributions near welds and IGSC crack growth rates, the current approach used by NRR is not considered unduly conservative for current BWR piping materials and environments. The Task Group endorses the guidelines for the evaluation of flaw growth in weldments given in draft NUREG-0313, Rev. 2.

- The Task Group also wishes to point out, however, that although conservative estimates of the throughwall growth of stress-corrosion cracks can be made with a high degree of confidence given an initial crack size, there is a high degree of uncertainty in the measurement of the depth of existing cracks, as discussed in Section 4, and this uncertainty must be adequately addressed in the development of acceptance criteria for flawed piping.

- The use of loading conditions defined by the ASME Code in performing flaw evaluations is acceptable, with the exception that SLA thermal expansion stresses should be included in the flaw evaluation when limit load is not achieved prior to significant crack extension. Use of these loading conditions is consistent with the approach which has been accepted by the NRC for the design and operation of other reactor components and systems.

- Analyses for large surface cracks, conducted using the relatively low ductile fracture resistance properties of stainless steel weld metal, show that the margins against fracture may not be as great as those believed to be associated with IWB-3640. Nonetheless, a substantial margin against fracture can be demonstrated for these flaws under normal operating conditions.

- Evaluation of throughwall cracks, using weld metal properties, shows that a surface crack breaking through the pipe wall under bounding accident loading conditions would have to be approximately 30% of the circumference in length, for large-diameter pipes, to result in unstable fracture of the pipe. However, leaks from cracks not large enough to cause gross failure of the pipe may be undesirably large.

- On the basis of fracture mechanics evaluation for bounding and typical stress conditions and weld toughness properties, it is concluded that IWB-3640 provides an adequate basis for evaluating the majority of the weld connections in BWR recirculation piping. This is especially true because many of the cracks will be in higher toughness zones adjacent to the lower toughness welds.

- Specific assessments of piping integrity will have to recognize weld conditions having lower toughness than the 304 stainless steel welds used in the analyses described in this report.

- Operating experience suggests that leak-before-break is the most likely mode of failure for the vast majority of cracks occurring in service. This is a result of the asymmetry of the weld residual stresses and applied loads and the variability in material properties. Evaluations using conservative crack growth rate predictions and net section collapse analyses, applicable to cracks in very high toughness material, indicate that for the vast majority of possible crack geometries there exists significant time to allow for detection of leakage and implementation of corrective actions. Evaluation using the fracture resistance properties of the weld material show substantial margins against failure, under normal and accident loading conditions for throughwall cracks which should be reliably detected by leakage.

Recommendations

Based on the above discussions and conclusions, the Task Group has developed the following recommendations.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair. The limitation on acceptable crack length is primarily a result of the lack of confidence in flaw depth sizing capability, and is intended to ensure leak-before-break conditions. The maximum allowable throughwall crack length can be determined based on weld joint specific loads.

- The maximum crack length allowable without repair for a specific weld joint should be the minimum of either 1) the throughwall crack length demonstrated by elastic-plastic fracture mechanics analyses to be stable under normal operating plus SSE loading conditions, 2) the throughwall crack length that would still permit the pipe to withstand normal operating plus SSE loading conditions as demonstrated by net section collapse (limit-load) analyses, or 3) the maximum crack length that would result in a leak rate greater than the plant's normal makeup capacity. Shorter cracks can be evaluated using the IWB-3640 criteria as modified by the NRC staff in SECY 83-267C. Calculations indicate that in the majority of cases the maximum crack length acceptable under the above criteria will be approximately 25% to 30% of the pipe circumference.

- The recommendation for a limit on acceptable crack length is primarily a result of the lack of confidence in ultrasonic depth sizing capabilities. In this respect, the fracture mechanics calculations presented in this report demonstrate that there are acceptable margins against fracture for relatively long, deep surface cracks. Demonstration of reliable sizing of the part-through flaw depth would most likely allow relaxation of the limits on crack length.

- Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair.

- Additional fracture mechanics analyses, material properties characterization, and large-scale pipe tests should be performed to further our understanding of the implications of stainless steel weld and cast material fracture toughness properties in flawed pipe evaluations. Furthermore, the Task Group recommends active NRC support of the ASME task group currently evaluating the concerns which have been raised regarding IWB-3640.

- Since operating experience and fracture mechanics evaluations indicate that leak-before-break is the most likely mode of piping failure, it is recommended that reasonably achievable leak detection procedures be in effect in operating plants. Current sump pump monitoring systems are sensitive enough to provide additional margin against leak-before-break if more stringent requirements on surveillance intervals and unidentified leakage are imposed (see Section 4). Therefore, the Task Group recommends that the limits on unidentified leakage in BWRs be decreased to 3 gpm and that the surveillance interval be decreased to 4 hours or less.

6.2 Short-Term Solutions

- For relatively short axial cracks, analysis can be used to justify long-term operation with weld overlays, since errors in crack depth measurement or flaw growth predictions for these cracks will lead at worst to relatively small leaks, which will be easily detectable long before the crack can grow long enough to cause failure. For circumferential cracks, weld overlay is considered an acceptable repair procedure for a maximum of two refueling outages unless reliable techniques for the sizing of cracks through the overlay or for the monitoring of crack growth are developed.

6.3 Long-Term Solutions and Safety Assessment

- Experience with materials to mitigate IGSCC, such as 347NG in Germany and 304NG and 316NG in Japan, has been excellent. Other materials used in the U. S. include 304L and 316L. All of these alloys are more resistant to IGSCC than conventional Types 304 or 316 stainless steel. Based on U. S. data and prior use, Type 316NG stainless steel offers an additional margin of resistance to IGSCC and utilities that choose to replace pipe should be strongly encouraged to use it.

- The use of additional mitigating procedures such as IHSI or hydrogen water chemistry to provide additional margin against environmentally assisted cracking is strongly recommended even after replacement with Type 316NG stainless steel. Utilities should be encouraged to adopt an ALARA approach to coolant impurity levels.

- Although low-carbon stainless steels with nitrogen additions have been successfully fabricated and welded in Japan and Europe, U. S. experience with these materials is limited. It appears that greater care must be exercised in the control of composition and fabrication variables to limit cracking during hot forming or welding.

- IHSI is considered to be a more effective mitigating action for IGSCC than HSW and LPHSW in part because more data are available to demonstrate that the process does produce a more favorable residual stress state. All the residual stress improvement remedies are considered to be more effective when applied to weldments with no reported cracking.

- IHSI is an acceptable mitigating procedure for weldments with no reported cracking even in older plants. However, to be fully effective a suitable hydrogen water chemistry also should be implemented.

- The use of IHSI on weldments with detectable cracking must be considered on a case-by-case basis. However, for relatively short cracks (approximately 20% of the circumference in length), since even large errors in crack sizing or the prediction of flaw growth will lead only to small leakage, the decision on whether an additional repair is required can be determined by analysis. For longer cracks, repair will probably be required.

- All steps related to the weld prep, welding, and weld finishing and HSW or IHSI should be validated through certification and through use of an appropriate QA manual. As noted previously, the weld should be optimized for future ISI.

- BWR water chemistry controls should be modified to minimize IGSCC. These modifications should include both a substantial reduction in the levels of ionic species entering the primary coolant and a control of oxygen level. The current work on reduction of oxygen through hydrogen additions should be followed closely with the possibility that it may be employed to reduce further the electrochemical potential of the stainless steel to a level at which SCC, either IGSCC or TGSCC, will not occur. It appears that hydrogen water chemistry is an effective IGSCC countermeasure. However, ongoing work regarding potential adverse effects on other reactor components should be closely followed in order to confirm the acceptability of this countermeasure.

- Either reduction of ionic species or hydrogen water chemistry, used alone, will have some beneficial effects. Used in combination, they show great promise for mitigating IGSCC.

- The effects of hydrogen water chemistry on the balance of the system and on overall long-term plant operation need to be explored in greater depth before they can be recommended without reservation for adoption on a wide scale.

- Regulatory Guide 1.56 needs to be modified to reflect these water chemistry recommendations to provide tighter controls on water conductivity and provisions for oxygen control as an acceptable option.

- The long-term effects of decontamination procedures on IGSCC need to be investigated.

9.0 IGSCC RISK STUDIES

Conclusions

- Intergranular stress-corrosion cracking (IGSCC) has caused no significant loss-of-coolant events or loss-of-coolant accidents (LOCA) in BWRs even though its presence in 304 stainless steel coolant piping has been found in many operating plants. This may be due to an aggressive in-service inspection and repair program and/or a tendency to leak-before-break (with effective leak detection). The LOCA contribution to the core damage frequency as calculated by probabilistic risk assessment (PRA) studies for BWRs is a minor contribution (generally less than 10%) for all LOCA sizes. Because the LOCA contribution to the core damage frequency is small, it would require a considerable error, at least an order of magnitude, from IGSCC before LOCAs would be a dominant contributor to the core damage frequency.

- IGSCC is but one component of the LOCA occurrence rate. Design and construction errors, maintenance errors, cyclic and thermal fatigue, etc. all contribute to the LOCA probability. It is estimated that IGSCC may be 30% or more of the LOCA frequency in BWRs as deduced from experience, the total industrial data base, and engineering judgment. Therefore, an error in the IGSCC LOCA contribution of 20-30 would be required for an order of magnitude error in the LOCA probability.

- The pipe failure data base and available studies were reviewed for evidence of IGSCC impact. To a considerable extent, uncertainties have always existed in determining piping failure rates. Largely, this uncertainty is intrinsic in attempts to measure low-frequency events from experience, and it makes the effect of trends difficult to analyze. It was statistically reasoned that an error of an order of magnitude in the large pipe failure frequencies (presumably from IGSCC) would have produced, with one chance in four, at least one failure during this interval. Therefore, it was concluded that the frequency for large pipe failure adequately accounts for the contribution due to IGSCC.

- For intermediate and small pipe failures, the incidence of reported and identified corrosion events in the data base, although increasing somewhat over time (owing to improved reporting and other causes), was judged to be not statistically indicative of an error in the failure frequencies.

- The systems aspects of IGSCC risk were revised using Hatch-2 as a model. It was concluded that the plants should be required to be designed for the most severe large LOCA without significant core damage. The ECCS must be able to survive the most severe single failure in the system and still perform its intended function(s). Pipe failure is no more severe than other failures in the system (valves, pumps, etc.) and is probably less severe. Coupling this with the ECCS redundancy and conservative margin in each design and calculational procedures, it is concluded that IGSCC would cause little increase in risk from the standpoint of systems important to safety.

- Multiple failures were briefly considered. IGSCC may be a common mode failure in the recirculation piping and other systems. It is concluded that a large LOCA and ECC system failure both caused by IGSCC is adequately protected by the single-failure criteria and systems redundancy. Multiple failures causing a LOCA larger than a DBA are likely to add little to the risk because it appears that the probability of multiple failures is several orders of magnitude less than that of a large single failure.

- Overall it is concluded that the presence of IGSCC in BWRs may reduce safety margins believed to exist, but there is no apparent evidence to conclude that IGSCC has increased public risk significantly.

Recommendations

Piping failure rates are difficult to determine because failures of high quality piping occur infrequently and require a data base acquired over a large number of reactor years, since normal experimental procedures are impractical. This difficulty is unlikely to be overcome now or in the foreseeable future. What results is an estimated frequency with large uncertainties from aggregated events acquired from poor-quality documentation. To partition this frequency according to contributing phenomena in a meaningful way is very difficult.

- It is recommended that a formal study and updating of the pipe failure data base be undertaken. Some statistics may be only meaningfully derived or assessed analytically. The NRC has developed a computer code PRAISE (Piping Reliability Analysis Including Seismic Events) which is currently being modified to include a model for IGSCC based on current understanding.

- It is recommended that PRAISE be used to a) investigate the impact of IGSCC in primary piping according to pipe size, b) study the failure frequencies of components due to IGSCC, fatigue, etc., and c) calculate the conditional probability of multiple failures associated with IGSCC to check the point value and to test the assumption of a lognormal distribution.

8.0 INTERGRANULAR STRESS-CORROSION CRACKING (IGSCC) VALUE-IMPACT ASSESSMENT

Conclusions

- The analysis indicates that this issue has low public risk impact, based on risk sensitivity studies and a review of pipe failure data including statistical estimates of IGSCC contribution to failure.

- Each of the scenarios has low value-impact ratios. However, when considering 30-35 affected BWR plants, the total potential public risk reduction indicates a low to at most medium safety priority ranking.

- However, economics dictate taking some action to reduce IGSCC impacts. IHSI with or without HWC, HWC alone, or pipe replacement are the order of cost preference. All options will reduce cost below the inspect and repair strategy alone. The results are not sensitive to discount rates except in the case of inspect and repair costs.

- HWC is not cost effective with pipe replacement and is marginal with IHSI. HWC in plants with existing cracking problems that prevent the use of IHSI is advantageous.

- ORE commitments for all strategies are large. IHSI will minimize the impact. Inspection and repair, even with HWC, will eventually exceed pipe replacement in man-rem.

APPENDIX A

LIST OF TASK GROUP MEMBERS

Spencer H. Bush (Chairman)	RSA	President Review and Synthesis Assoc. Richland, WA
Richard A. Becker	NRC	Office of Nuclear Reactor Regulation Clinch River Breeder Reactor Program Office Technical Review Section
Ching-Yao Cheng	NRC	Office of Nuclear Reactor Regulation Division of Engineering Materials Engineering Branch
William J. Collins	NRC	Engineering & Generic Communications Branch Office of Inspection & Enforcement
Billy R. Crowley	NRC	Engineering and Operations Program Division Regional Administrators Office, Region II
Jacque P. Durr	NRC	Regional Administrators Office, Region I
Warren S. Hazelton	NRC	Office of Nuclear Reactor Regulation Division of Engineering Materials Engineering Branch
Philip R. Matthews	NRC	Office of Nuclear Reactor Regulation Division of Safety Technology Safety Program Evaluation Branch
Joseph Muscara	NRC	Office of Nuclear Regulatory Research Division of Engineering Technology Materials Engineering Branch
Richard C. Robinson	NRC	Office of Nuclear Regulatory Research Division of Risk Analysis Reactor Risk Branch
Jack Strosnider	NRC	Office of Nuclear Regulatory Research Division of Engineering Technology Materials Engineering Branch

APPENDIX B

LIST OF CONSULTANTS AND MAJOR CONTRIBUTORS

William Andrews	Battelle Memorial Institute Pacific Northwest Laboratories	Value-Impact
Steven R. Doctor	Battelle Memorial Institute Pacific Northwest Laboratories	Nondestructive Examination
Ronald M. Gamble	IMPELL Corporation	Fracture Mechanics
David Kupperman	Argonne National Laboratory	Nondestructive Examination
William J. Shack	Argonne National Laboratory	Materials Engineering
Tom T. Taylor	Battelle Memorial Institute Pacific Northwest Laboratories	Nondestructive Examination
John R. Weeks	Brookhaven National Laboratory	Corrosion Mechanisms and Water Chemistry
Gery Wilkowski	Battelle Columbus Laboratories	Fracture Mechanics

APPENDIX C

This appendix consists of two parts. The first, C-1 is a copy of the 10 CFR 50.54 position pertaining to inspection measurements. Basically, the same information was contained in SECY-83-267c.

The second, C-2, is a copy of the 10 CFR 50.59 requirements pertaining to the replacement of piping in BWRs.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
April 19, 1984

TO ALL LICENSEES OF OPERATING REACTORS, APPLICANTS FOR OPERATING LICENSE, AND HOLDERS OF CONSTRUCTION PERMITS FOR BOILING WATER REACTORS

Gentlemen:

SUBJECT: INSPECTIONS OF BWR STAINLESS STEEL PIPING (Generic Letter 84-11)

Inspections conducted at several boiling water reactors (BWRs) revealed intergranular stress corrosion cracking (IGSCC) in large-diameter recirculation and residual heat removal piping. These inspections were conducted pursuant to IE Bulletins 82-03, Revision 1 and 83-02, and the NRC August 26, 1983 Orders. The Commission believes that the results of these inspections mandate an ongoing program for similar reinspections at all operating BWRs. Where IGSCC is discovered, repairs, analysis and additional surveillance may also be required to ensure the continued integrity of affected pipes.

Staff efforts to date on this issue include review of the Electric Power Research Institute (EPRI) report dated August 4, 1983, establishment of a pipe crack study group within the staff, evaluation of the results of IGSCC inspections already conducted, and discussions with licensees and industry groups. As a result of these considerations, the staff has concluded that the following actions would be considered an acceptable response to the current IGSCC concerns:

1. A reinspection program of piping susceptible to IGSCC should be undertaken. The reinspection should commence within about two calendar years, adjusted to coincide with the next scheduled outage, from the previous inspection performed under IE Bulletins 82-03, 83-02, or our August 26, 1983 Order.

2. These reinspections should include the following stainless steel welds, susceptible to IGSCC, in piping equal to or greater than 4" in diameter, in systems operating over 200°F, that are part of or connected to the reactor coolant pressure boundary, out to the second isolation valve as follows.

   (a) Inspection of 20% of the welds in each pipe size of IGSCC sensitive welds not inspected previously (but no less than 4 welds) and reinspection of 20% of the welds in each pipe size inspected previously (but not less than 2 welds) and found not to be cracked. This sample should be selected primarily from weld locations shown by experience to have the highest propensity for cracking.

   (b) All unrepaired cracked welds.

   (c) Inspection of all weld overlays on welds where circumferential cracks longer than 10% of circumference were measured. Disposition of any findings will be reviewed on a case-by-case basis. Criteria for operation beyond one cycle with overlaid joints are under development.

(d) Inspection of any weld treated by induction heating stress improvement which has not been post treatment UT acceptance tested.

(e) In the event new cracks or significant growth of old cracks* are found, the inspection scope should be expanded in accordance with IEB 83-02.

NOTE: [Results of inspections conducted to date under IEB 82-03 and 83-02 indicate that all stainless steel piping welds in systems operating over 200°F are susceptible to IGSCC. In addition, field data shows that the cracking experience does not correlate well with the Stress Rule Index (SRI) and the carbon content. Therefore, the primary index for sample selection should be field experience, where other factors such as weld preparation, excessive grinding, extensive repairs, or high stress locations are known to exist, they should also be considered in the sample selection.]

3. All level 2 and level 3 UT examiners should demonstrate competence in accordance with IEB 83-02 and level 1 examiners should demonstrate field performance capability.

4. Leak detection and leakage limits should be sufficiently restrictive to ensure timely investigation of unidentified leakage. See Attachment 1.

5. For crack evaluation and repair criteria see Attachment 2.

Accordingly, pursuant to 10 CFR 50.54(f), BWR operating reactor licensees and applicants for an operating license (this letter is for information only for those utilities that have not applied for an operating license) are requested, in order to determine whether your license should be modified or suspended, to furnish, under oath or affirmation, no later than 45 days from the date of this letter, your current plans relative to inspections for IGSCC and interim leakage detection. Your response should indicate whether you intend to follow the above staff recommended actions or to propose an alternative approach to resolving IGSCC concerns. In either case, your response should address:

(a) Scope and schedule of planned inspections

(b) Availability and qualification of examiners

(c) Description of any special surveillance measures, in effect or proposed, for primary system leak detection, beyond those measures already required by your Technical Specifications

(d) Results of the Bulletin inspections not previously submitted to NRC

(e) Remedial measures, if any, to be taken when cracks are discovered

*Significant growth of the old crack is defined as growth to a new crack size that cannot be accepted without repair for the remaining period of the current or a new cycle of operation, in accordance with Attachment 2.

The staff considers the IGSCC problem to be generic for all BWRs. Therefore, your response may incorporate by reference materials furnished by an Owners' Group. To the extent practicable, Owners' Group and EPRI participation in the IGSCC effort is encouraged.

Licensees and applicants may request an extension of time for submittals of the required information. Such a request must set forth a proposed schedule and justification for the delay. Such a request shall be directed to the Director, Division of Licensing, NRR. Any such request must be submitted no later than 15 days from the date of this letter.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0011 which expires April 30, 1985. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D. C. 20503.

Darrell G. Eisenhut, Director
Division of Licensing

ATTACHMENT 1

LEAK DETECTION AND LEAKAGE LIMITS

The reactor coolant leakage detection systems are operated in accordance with the Technical Specification requirements to assure the discovery of unidentified leakage that may be caused by throughwall cracks developed in austenitic stainless steel piping.

A. The leakage detection system shall be sufficiently sensitive to detect and measure small leaks in a timely manner and to identify the leakage sources within practical limits. Particular attention should be given to upgrading and calibrating those leak detection systems that will provide prompt indication of an increase in leakage rates

Other equivalent and/or local leakage detection and collection systems will be reviewed on a case-by-case basis.

B. Plant shutdown shall be initiated for inspection and corrective action when any leakage detection system indicates, within any period of 24 hours, an increase in rate of unidentified leakage in excess of 2 gpm or its equivalent, whichever occurs first. For sump level monitoring systems with a fixed-measurement interval method, the level shall be monitored at 4-hour intervals or less.

C. At least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours or immediately initiate an orderly shutdown.

D. Unidentified leakage should include all leakage other than

(1) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or

(2) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operations of unidentified leakage monitoring systems, or not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

E. A visual examination for leakage of the reactor coolant piping shall be performed during each plant outage in which the containment is deinerted. The examination will be performed consistent with the requirements of IWA-5241 and IWA-5242 of the 1980 Edition of Section XI of the ASME Boiler and Vessel Code. The system boundary subject to this examination shall be in accordance with IWA-5221.

ATTACHMENT 2

CRACK EVALUATION AND REPAIR CRITERIA

1. Background

(a) Code Requirements

The ASME Boiler and Pressure Vessel Code Section XI has rules for evaluating the acceptability of flaws for further operation. Table IWB 3514-3 provides rules for acceptability of flaws without further evaluation; although the specific dimensions of such acceptable flaws depends on both the length and depth of the flaws, the practical effect is that flaws less than about 10% of the wall thickness are acceptable for further operation without analysis or repair.

A new section has recently been added to the Code, IWB 3600. This extends the Code flaw evaluation rules for piping to include specific rules whereby flaws deeper than those allowed by IWB 3514-3 can be accepted for further operation without repair.

Section IWB 3600 also requires that these acceptable flaw sizes include considerations of crack growth by stress corrosion and fatigue. In other words, if a crack is to be considered acceptable for further operation without repair, it must be shown that it will not grow to be larger than the IWB 3640 limits during the time period for which the evaluation is performed.

(b) Crack Growth Assessment

IGSCC at welds in BWRs is primarily initiated by very high tensile residual welding stresses on the inside surface at the heat affected (sensitized) zones of the base metal very near the welds. This tensile residual stress changes to a compressive stress toward the middle of the pipe wall; this reduction in stress reduces the crack growth rate through the center portion of its pipe wall. As the crack progresses further through the wall, the relative effect of the pressure and bending stresses increases, and the crack growth rate will increase.

The residual stress patterns and calculational methods for crack growth rates are fairly well established by considerable research and correlations with service experience. The staff has selected parameters that should lead to overprediction of growth. This is intended to compensate for uncertainties discussed in more detail below.

(c) Staff Treatment of Uncertainties

One of the main uncertainties associated with the evaluation of pipe cracks is the uncertainty of crack sizing, both depth and length of IGSCC cracks. Although this technology is being improved, the uncertaintiy in crack sizing will likely remain.

The staff has used a relatively simple approach to cover sizing uncertainties. In practice, the staff approach permits operation of unrepaired cracks but only if calculations show that they would not exceed Code limits even if the crack at the start of operation were actually twice as large as reported.

2. Staff Acceptance Criteria

(a) Criterion for Operation without Repair

Plant operation is permitted with cracked welds only for the time period that the cracks are evaluated to not exceed 2/3* of the limits for depth and length provided in ASME Code Section XI, Paragraph IWB-3640. Crack growth analyses must include any additional stress imposed on the weld by other weld repair operations, and each analysis must be approved by the NRC.

(b) Criteria for Cracked Repairs

(i) If cracked welds are repaired by weld overlay, the thickness of the overlay must be sufficient to provide full IWB-3640 margin during the proposed operating period, assuming that the cracks are or will grow completely through the original pipe wall and the first overlay layer to the low carbon and low ferrite portion of the overlay, unless it is demonstrated that the crack(s) are shallow enough to be arrested by the weld overlay.

*This criterion allows for an uncertainity of up to 100% in crack depth siz for reported cracks up to 25% of wall thickness.

Effective overlay thickness is defined as the thickness of overlay deposited after the first weld layer that clears dye-penetrant testing (PT) inspection.

(ii) The minimum effective overlay thickness permitted, even for very short cracks in either longitudinal or circumferential direction, is two weld layers after the first layer to clear PT inspection.

(iii) Full structural strength weld overlays must be provided for long cracks with total circumferential extent approaching the length that would cause limit load failure if they were actually through-wall.

(iv) Multiple short circumferential cracks are to be treated as one crack with a length equal to the sum of the circumferential lengths.

3. Discussion of Staff Acceptance Criteria

Since the period of operation between inspections could vary from plant to plant and the applied stress level varies from location to location, use of a fixed simplified repair criterion established on the bases of crack size prior to the period of operation would be difficult. In any case, however, flaws less than about 10% of the wall thickness are acceptable for further operation without repairs. For a typical 18 month operating cycle, the staff criteria would generally require that cracks greater than 30% of the circumference and cracks with reported depth of 25% or greater of the thickness will likely need some form of repair. For the same 18 month cycle, cracks of smaller size down to 10% of wall thickness may be acceptable without repair but would require evaluation in accordance with the staff criteria.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 14, 1984

TO ALL LICENSEES OF BOILING WATER REACTORS (BWRS)

Gentlemen:

SUBJECT: PROCEDURAL GUIDANCE FOR PIPE REPLACEMENT AT BWRS (Generic Letter 84-07)

This letter provides guidance to licensees planning to replace recirculation system piping (or other reactor coolant system pressure boundary piping) with material that is less susceptible to intergranular stress corrosion cracking. In particular, guidance is provided regarding NRC reviews and approvals that may be necessary.

10 CFR 50.59 specifies the conditions that would require prior NRC approval of changes in the facility. In your compliance with 10 CFR 50.59, we recognize that the decision on whether your planned replacement program involves an unreviewed safety questions can be difficult, and that an understanding of the NRC position on this issue would be helpful in planning your program. The purpose of this letter is to provide as clear a statement as possible of our views on this issue.

We encourage programs to replace piping so as to minimize the potential for cracking and we will expeditiously review any submittals provided to us so as to not delay this important improvement program. We encourage early submittal of appropriate requests for review for those situations that require prior approval. Prior NRC approval is not necessary unless the proposed change to the facility involves an unreviewed safety question or a change in Technical Specifications.

In all cases, licensees must perform and document appropriate reviews and analyses in accordance with 10 CFR 50.59 and the facility Technical Specifications. These analyses should be maintained by the licensee, in accordance with Commission regulations and the applicable license, to permit the staff to audit such evaluations, as necessary. In those cases where licensees determine that their program for pipe replacement does not involve an unreviewed safety question, there remains the concern that the NRC may, at a later date, disagree with that determination, thereby potentially delaying the program. To minimize that possibility, we have developed a position regarding the major considerations in a pipe replacement program which licensees can use in determining the necessity or desirability of seeking prior NRC approval. That position is contained in the enclosure to this letter.

Replacement of recirculating system piping may involve individual and collective radiation exposure to plant workers beyond that in other routine maintenance work. 10 CFR Part 20 requires that licensees "make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable." We request that a

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description of your radiation protection program for the pipe replacement effort be furnished to us as early as possible before initiating the replacement program. Your submittal should include a description of appropriate pre-planning procedures, shielding, equipment, personnel training, estimated total cumulative dose, and other measures to be initiated that will keep exposures as low as reasonably achievable. We anticipate that most pipe replacment programs can be accomplished through suitable controls so as to limit cumulative exposures to less than about 2000 person-rem. We will plan to meet with licensees whose programs involve greater dose estimates.

This request has been approved by OMB Clearance Number 3150-0011, which expires April 30, 1985.

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Procedural Guidance

ENCLOSURE

PROCEDURAL GUIDANCE FOR PIPE REPLACEMENT AT BWR'S

As a result of the occurrence of intergranular stress corrosion cracking (IGSCC) in the recirculation system piping at a number of Boiling Water Reactors (BWRs), several licensees have indicated their intention to replace the recirculation piping with material that is less susceptible to IGSCC. The purpose of this document is to present NRC staff guidance regarding procedural requirements applicable to the replacement of recirculation system piping. Depending on the specific changes that may be involved, such changes may be authorized in various ways under Commission regulations.

Replacement of a reactor coolant system component by another component of the same kind (not changing any feature described in the FSAR) would not be prohibited by the Commission regulations.

Replacement which involves changes, but which does not involve an amendment to the license or Technical Specifications, may be carried out without prior Commission approval, if the changes involve no unreviewed safety questions. The definition of an unreviewed safety question is set forth in 10 CFR 50.59. Most Technical Specifications contain provisions governing the way a licensee is to assess changes to determine whether they involve unreviewed safety questions. Not only should each aspect of change be carefully considered in making a determination under 10 CFR 50.59, but the overall cumulative effect of the various changes, considered as a whole, should also be assessed.

There may be cases involving difficult questions concerning whether some aspect may be an unreviewed safety question. Such cases may be processed as a license amendment authorizing the replacement program*. Changes which entail changes in Technical Specifications may be authorized only by license amendment**. License amendments involving no significant hazards considerations will be processed expeditiously - after 30 days notice in the Federal Register and sufficient time for staff review.

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*The case of Nine Mile Point, the first major pipe replacement case, was handled by the amendment process.

**If the changes in Technical Specifications are limited in number and can be evaluated as a matter distinct from the replacement program, such changes may be processed separately from the replacement program.

If there is a request for a hearing concerning an amendment for which there is a final determination that the amendment involves no significant hazards, the amendment may be granted notwithstanding such request. The hearing, if any, would be held after the amendment has been issued. If the changes are such as to involve a significant hazards consideration, and a hearing is requested, the license amendment would await the outcome of the proceeding.

We have reviewed information concerning the programs under consideration by some licensees, and conclude that certain proposed features of such programs having the characterisitics discussed below, subject to the results of the plant specific evaluations required by 10 CFR 50.59, do not involve unreviewed safety questions.

Design Specifications

Design specifications are prepared covering all aspects of construction of the replaced piping system, including the Codes designated for use. Construction as used above is an all-inclusive term comprising materials, design, fabrication, installation, examination, testing and inspection.

Codes

Either:

(1) The codes and standards described in the FSAR, consistent with current regulations, as the original licensing basis for issuance of the facility operating license are utilitized;

Or, and preferably:

(2) The Code edition used for the construction of the piping system is, to the extent practical, the latest Code edition and addenda. If the later Codes are used, practical adjustments are made in using upgraded Code provisions to accommodate the limitations of design, geometry and items of the original piping systems and its supports which are not replaced. The quality and overall margins required in the original design are not to be impaired in determining the extent to which adjustments are made in using upgraded Code provisions.

Materials

The materials used are those that have received staff reveiw and approval as documented in NUREG-0313, Rev. 1.

Design Changes

Replacement components and their resulting effect upon system performance provide an equivalent overall safety margin as provided in the originally licensed design. Within this constraint, there may be design changes that clearly enhance the safety function of the system under replacement, such as, changes to reduce the number of welds or changes to facilitate inspection. Piping, pipe supports and any remaining original components are covered in the specifications of loading combinations and appropriate allowable limits provided for the replaced system.

Analysis

A stress analysis for the recirculation and other replaced piping systems is to be performed, that demonstrate that allowable limits have not been exceeded. The loads and loading combinations used in the FSAR or as described in SRP Section 3.9.3 should be used in the stress analysis.

Whip restraints and jet impingement design must follow FSAR criteria or that of Standard Review Plan Section 3.6.2.

Fabrication, Installation, Examination and Testing

Fabrication, installation, examination and testing is to be performed in accordance with the applicable Code designated under Codes.

Inspection/Quality Assurance

Programmatic quality assurance and independent (third party) inspection requirements appropriate to the replaced piping are to be followed.

System Characteristics

Flow rates, temperatures, and pressures must not be significantly different than those evaluated in the FSAR. The performance of Engineered Safety Features must not be degraded by the replaced piping systems.

Discussion of 10 CFR 50.59

The NRC staff has developed this guidance without incorporating the results of any detailed review of a specific plant or specific plant Technical Specifications. A detailed case-specific review could lead to a positive unreviewed safety question conclusion even though the general guidance provided above would suggest a negative conclusion. A positive conclusion would be reached if, for example, the combination of FSAR codes and updated codes leads to a reduced safety margin for some plant structure, system or component, and this reduced margin either increases the possibilities or consequences of an accident or malfunction of equipment important to safety which was evaluated in the FSAR, or created the possibility of some new accident or malfunction, or pertains to a margin of safety in the basis for any Technical Specification. The margin of safety defined by a Technical Specification would comprise several safety margins within a given system. In such a circumstance the margins of safety for individual components within the system could well be adjusted without any effect on the margin of safety for the system as defined in the Technical Specifications. Thus each licensee must perform the specific, detailed review required by 10 CFR 50.59(a)(2) and in all cases the results of the detailed, case-specific review are controlling over the general guidance provided in this enclosure. Not all repairs involve changes and if they do not, no plant-specific review pursuant to 50.59 is required.

APPENDIX D

INTERGRANULAR STRESS-CORROSION CRACKING RISK STUDIES

D.1 INTRODUCTION

Early efforts concerning the impact of IGSCC on nuclear power plant safety relied on the application of a deterministic approach, the established licensing methodology. In general, this procedure utilizes the established engineering sciences to form engineering judgments regarding the impact of a phenomenon on safety margins.

Some degree of quantification of unlikely events through probabilistic assessments did not come fully into the regulatory process until the Reactor Safety Study (D.1) was published in late 1975. Several controversial years followed before probabilistic risk assessment (PRA) techniques were accepted and applied in a fairly routine manner. Acceptance does not imply total reliance or that the uncertainties in the methodology are ignored, but PRA is generally accepted as a useful tool. Currently, a number of PRAs have been performed on operating plants either as full-scope PRAs or through the Interim Reliability Evaluation Program (IREP).

The mounting concern, as more IGSCC is detected by BWR inservice inspections, is discussed elsewhere in this report. More frequently, inservice inspections have discovered IGSCC in the large-diameter recirculation piping both in domestic and foreign BWRs. As more piping cracks attributable to

IGSCC are discovered, the question of IGSCC implications to public risk logically arises. One facet of the answer to the implications of IGSCC to public risk may be addressed by examining the implications inferred from existing risk assessments. The remainder of this Appendix examines these risk implications as they relate to BWRs only.

D.1.1 Loss-of-Coolant Accident (LOCA)

All current BWRs operate on a direct cycle: i.e., water removes the heat from the nuclear fuel by boiling in the core, the steam and water are separated in the vessel, the steam is taken directly from the vessel to drive the turbine-generator, and the remaining water is recirculated through the core. If the primary pressure boundary fails, a LOCA results. The failure of a pipe, weld, or boundary component results in a LOCA. It has been common practice (D.1) to define LOCA accidents in ranges of rupture sizes. Nominally, the rupture sizes correspond to area of the opening, but have been given equivalent piping diameters. This categorization for BWR liquid piping* is 8-1/2 in. and greater, between 8-1/2 and 2-1/2 in., and between 2-1/2 and 1/2 in. This structuring is made loosely on system considerations which will be discussed more fully in a later section. In LWRs this structuring defines large LOCA b(A), small LOCA b(S_1), and small-small LOCA b(S_2) sequences.

*WASH-1400 defines different ranges for steam piping, but these variations are unimportant for this discussion. In reading WASH-1400, confusion between the two sets of limits can result since the suction side of the RCS is liquid and the discharge side is steam.

D.1.2 Risk Assessment Methodology

PRA methodology attempts to examine all the accident sequences which could potentially endanger the general public. By quantifying the likelihood of these sequences and the potential consequences should they occur, the overall risk to public safety and health may be estimated. Although there are other sequences, the dominant contributors to public risk are those accident sequences that lead to reactor core damage and, consequently, release of radioactivity to the surroundings. Since the release of radioactivity after core damage is common to most sequences, the relative contribution to public risk is assumed in this study to be proportional to the contribution an accident sequence makes to the core damage frequency.

The first question to be addressed is: What is the contribution of LOCA sequences to the core damage frequency? Table D.1 presents the results taken from a number of BWR PRAs, including WASH-1400. It can be seen that the LOCA contribution in all cases is a small fraction of the core damage frequency. With the detection of more IGSCC in BWR piping, the question of impact on risk focuses on the impact IGSCC has on failure rate. Before proceeding further, a sensitivity model to assess the impact of the failure rate on core damage frequency proportionate to risk is necessary. A simple sensitivity model can be developed from the risk methodology.

Table D.1

Sensitivity of Core Melt Frequency for BWR PRAs
for a Postulated Increase (10x) in LOCA Initiator Frequency

Plant A*	Sponsor	Total Core Melt Frequency** (per Rx yr)	Frequency of Core Melt Due to LOCA Initiated Sequence (per Rx yr)	Increase in the Frequency of Core Melt Due to Postulated Increase (10x) in LOCA Initiator Frequency (per Rx yr)
Limerick (Rev 5)	PECO.	1.5 E-5	1.3 E-7	1.2 E-6
Shoreham	LILCO	5.5 E-5***	1.9 E-6	1.7 E-5
Millstone 1	IREP	3.0 E-4	3.0 E-6	2.7 E-5
Browns Ferry	IREP	2.0 E-4	1.2 E-6	1.2 E-5
Peach Bottom (RSS)	NRC	2.9 E-5	1.2 E-6	1.1 E-5

*Big Rock Point was considered but deemed atypical, since it is a BWR-1 and has a less comprehensive ECCS. GESSAR (Grand Gulf) was also considered, but was judged inappropriate for this list because it is a BWR-6, among other reasons.

**Total core melt frequencies reported in the probabilistic analyses used for this study resulted from internal initiators only. Analyses including external initiators would yield higher core melt frequencies and, thus, a lower percentage contribution from LOCAs.

***The Shoreham value for "core vulnerable frequency" is believed to be representative of core melt frequency and has been used here.

D.2 SENSITIVITY MODEL

As described in WASH-1400 (p. 166, Main Report), the LOCA accident sequence probability is given by

$$P_{IE} \times P_{SF} \times P_{CFM} \times P_{WC} \times P_{PD} \quad , \qquad (1)$$

where IE is the initiating event; SF is system failure; CFM is containment failure model; WC is weather conditions; and PD is population density. Because (1) is a simple linear product, the only term which is germane is P_{IE}, which in the case under consideration is the frequency of pipe failure initiating the LOCA. P_{IE} may be expressed as

$$P_{IE} = 1 - R_P \quad , \qquad (2)$$

where Rp is the reliability of the plant piping. If the plant piping is considered simply as characterized by a failure rate λ, (2) can be written

$$P_{IE} = 1 - e^{-\lambda t} \approx \lambda t \quad , \qquad (3)$$

assuming λt is small.

To a first approximation, the LOCA probability is directly proportional to the failure rate of the reactor coolant system (RCS) boundary. Assuming that all other factors are constant, the risk is then linearly proportional to the RCS failure probability. As noted earlier, it is sufficient to consider contributions up through the core damage frequency without losing generality. The λ of (3) does not represent the precise failure rate of the RCS piping for several reasons. First, the lack of rigor inherent in representing the entire reactor complex as a single simple component introduces approximations. Second, the failure rate is a function of failure definition, which is another way of describing partitioning by break size as discussed earlier. Neither of these concerns would negate the use of Eq. (3) as a sensitivity model or a parameter to examine the data base from which it was derived. Utilizing linear proportionality, it can be noted in the last column of Table D.1 that increasing the LOCA frequency by a factor of 10 is required in most cases, before the LOCA contribution is a significant fraction of the total core melt frequency.

An NRC internal study (D.2) of a similar sensitivity nature, which also considered transient induced LOCA, multiple pipe breaks, and seismically initiated LOCA, concluded that significantly large increases in frequency of pipe failure would be required before LOCAs would be dominant contributors to risk.

It can be concluded that: (a) the LOCA contribution in BWRs has not been found to be a dominant contributor to public risk in PRAs performed to date, and (b) that a substantial error in the frequency of pipe failures would be necessary before LOCAs could be considered a substantial contributor to public risk. These conclusions may vary on a plant-specific basis; however, current PRAs (like Grand Gulf) indicate that there is considerable conservatism in the LOCA contribution to risk in BWRs. The next step is to evaluate these failure rates and the data base(s) from which they are derived.

D.3 LOCA FREQUENCIES

The failure rates in current use for most studies appear to stem from the data base treatments in the RSS, WASH-1400, where the LOCA failure rate was partitioned into three contributions, apparently dictated by the mitigating capabilities of the Engineered Safety Features (ESF). This partitioning of the failure rate may be somewhat artificial with respect to the data base, but similar partitioning has been followed in most of the studies to date. The WASH-1400 partitioning is shown in Table D.2. The range can be viewed as associated with the uncertainty in the parameter. The uncertainty is large at the 90% confidence level, at least two orders of magnitude for each LOCA class.

D.3.1 Large LOCA

The failure frequency for large pipes resulting in a large LOCA is perhaps the most understandable of the LOCA failure frequencies because it represents large openings in the primary pressure boundary up to and including double-ended guillotine rupture of pipes of 8-1/2 in. equivalent diameter or larger. The data bases of references D.1 and D.3, in addition to the License Event Reporting (LER) system regarding piping breaks, were reviewed. No ruptures of this magnitude have occurred in the primary system in the approximately 700 LWR years amassed through November 1983 for the U. S. commercial nuclear power industry. A careful use of definitions is necessary here. First, there have been large pipe failures in the secondary system of PWRs as noted in reference D.3. However, the effective opening of the failure must correspond to an equivalent 8-1/2 in. pipe size to be consistent with the treatment in reference D.1, not simply any failure in a pipe of that size.

Table D.2**

RSS Pipe Failure Assessed Values

WASH-1900 Pipe Rupture Size (inches)*	LOCA Initiating Rupture Rates for Plant Per Year 90% Range	Median
1/2 - 2 1/2	$1x10^{-4}$ - $1x10^{-2}$	$1x10^{-3}$
2 1/2 - 6	$3x10^{-5}$ - $3x10^{-3}$	$3x10^{-4}$
>6	$1x10^{-5}$ $1x10^{-3}$	$1x10^{-4}$

*Rupture size area equivalent to pipe area of this diameter.
**Reference D.1.

When the RSS was performed, 17 reactors were selected for the collection of failure data during one year of operation (1972). Since 17 reactor-years of operational experience do not provide an adequate data base, a pooled data approach was adopted which utilized data bases from other nuclear operations, foreign nuclear operations, and diverse industrial experience. [See references D.1 and D.3 for comprehensive discussions of this procedure and the resulting failure rate statistics.] There was no attempt, and probably no need, to partition the relative contribution by cause or differentiate by explicit reactor type within the water-cooled reactor category.

For the demands of the RSS, the pioneering nature of its objectives and its relation to the existing experience base, these procedures were appropriate. However, treatment for LOCAs has been used with essentially no modification in the BWR PRAs reviewed. This is true for LWR PRAs in general (see reference D.4). Pooled data have the advantage of providing an estimated failure frequency with less statistical uncertainty. Pooled data have the disadvantages of contaminating the data base by mixing samples from disparate populations and the inability to partition the resulting failure rates by reactor type and cause. The cross-contaminants appear to be of roughly two broad and overlapping categories: (1) Phenomenologically induced failures predominantly associated with distinct populations, e.g., IGSCC in BWRs and other corrosive processes in the chemical industry; and (2) differing quality control and assurance standards. Both the advantage and disadvantages are significant problems in their own right and are not directly related to the objectives of this study. Discussions of data base concerns are in many of the cited references, but references D.1, D.3, D.5, and D.6 present more direct discussions of these problems.

The PCSG was confronted with the problem of the existing PRA methodology (pooled data) and its relevance to the population of completed PRAs opposed to IGSCC and its occurrence in BWRs. The following argument relating to a postulated error in the large pipe failure rate is predicated on 273 reactor years of U. S. commercial BWR experience through November of 1983. This approach has been adopted to avoid any basically irrelevant arguments about the data base quality.

An estimate of failure rate can be calculated by

$$\hat{\lambda} = \frac{r}{T} , \qquad (4)$$

where $\hat{\lambda}$ is the estimate of the failure rate or point value, r is the number of failures, and T is the sum of the operating times accumulated by all units. Where there are no failures, a realistic point estimate of the frequency may be obtained from

$$\hat{\lambda} = \frac{\chi^2_{0.50;\ 2r + 1}}{2T} \tag{5}$$

It can be shown that the unknown failure rate lies in the interval

$$\frac{\chi^2_{\alpha/2;\ 2r}}{2T} \leq \lambda \leq \frac{\chi^2_{1-\alpha/2;\ 2r + 2}}{2T}, \tag{6}$$

which has a confidence level of $1-\alpha$. It must be noted that Eq. (6) does not establish what the true failure rate is, only that it lies in the defined interval with probability $1-\alpha$. The best estimate of the failure frequency is the point value calculated by Eq. (4). It may also be shown that the median value of the chi-squared interval may be calculated by Eq. (5) regardless of the number of failures. Both the point and median value results have been included in Table D.3. Both methods produce comparable estimates. Equation (6) is calculated in Table D.3 for 7 hypothetical number of failures.

The apparent increase in IGSCC in BWR piping raises the question: Is it likely that the failure frequency might be higher by a statistically substantial amount than the value currently used for large LOCA failure rates? From Table D.3, the implication can be drawn that if the failure frequency were in error, by an order of magnitude, several failures would be expected to have occurred. This might lead one to conclude, therefore, that there is little significant evidence that an error of that magnitude exists in the failure frequency regardless of the ostensible cause. However, there exists the (statistical) possibility that the failure rate could be in error by an order of

Table D.3

Hypothetical Failure Frequency Estimates*
(Total U. S. BWR Operating Time (T) = 273 Reactor-Years)
95% Confidence Limits

pothetical* bserved" ilures (r)	Chi-Squared Table Value L	U	Failure Frequency Point Value (10^{-3})	Chi-Squared Confidence Limits		Chi-Squared Median (10^{-3})
0	0	7.38	0	0	1.35×10^{-2}	0.84
1	0.05	11.1	3.7	9.15×10^{-3} -	1.4×10^{-2}	4.4
2	0.48	14.4	7.4	8.8×10^{-4} -	2.6×10^{-2}	8.0
3	1.24	17.5	11.1	2.2×10^{-3} -	3.2×10^{-2}	11.6
4	2.18	20.5	14.8	4.0×10^{-3} -	3.8×10^{-2}	15.2
8	6.91	31.5	29.6	1.3×10^{-2} -	5.3×10^{-2}	29.8
10	9.59	36.8	37.0	1.8×10^{-2} -	6.7×10^{-2}	37.2

: $\chi^2_{0.025;\ 2r}$

: $\chi^2_{0.975;\ 2r + 2}$

:call, no large-pipe LOCA have been reported for primary systems.

magnitude and yet no failures have been observed in the interval T. This can be seen by the use of standard reliability concepts. The reliability of a component represents the probability that there will be no failure during the interval T.

If the reliability of a component population n is given by

$$R(n) = e^{-\lambda_n T} . \qquad (7)$$

The ratio of two component populations is then given by

$$\frac{R(1)}{R(2)} = e^{(\lambda_2 - \lambda_1)T} . \qquad (8)$$

If $\lambda_2 = 10\lambda_1$, Eq. (8) reduced to

$$\frac{R(1)}{R(2)} = e^{9\lambda_1 T} . \qquad (9)$$

Assume that T = 273 reactor years, and $\lambda_1 = 10^{-3}$; then

$$\frac{R(1)}{R(2)} \cong e^{2.46} \cong 12 .$$

However, if the total number of reactor years in the U. S. comercial program are used (representing pooled U. S. data), it is 500 times less likely to produce 700 failure-free years of operation for an order of magnitude underprediction error in the failure rate. Further, if worldwide reactor data are pooled (representing approximately 1000 failure-free operational years), it becomes even more unlikely. It can be noted that this argument is sensitive to the absolute value of the failure rate. However, the thrust of the argument is that there is little statistical support for the hypothesis that there is a significant error in the failure rate even though the presence of IGSCC may point to a greater failure precursor population. Statistically, it has already been pointed out that the true unknown failure rate lies in a wide uncertainty band, in this case at least two orders of magnitude in length.

What is really being asked is whether there is any evidence that there is a significant difference in the medians of the two populations. Since there have been no failures of large pipes, it can be statistically concluded that there appears to be no evidence of any difference, at least in a bounding sense.

It might be tempting to conclude that, since the frequency of large LOCA failures does not appear to be increasing, IGSCC has not been an over-looked factor in large LOCA failure rates. However, there are no data to support this conclusion directly since in-service inspection and repair/replacement may actually be responsible for avoiding large LOCA failures and reducing the failure rate (reference D.7, pp. 199-200).

Although more reactor years of operation continue to be amassed, it would be improper to conclude much beyond that the failure rate is in about the correct range consistent with the data. The failure rate could be much lower, for example, but the physical situation would appear to impose bounds at about 1×10^{-4} failures per year as a minimum failure rate that could be statistically justified on the basis of the following bounding argument. Assume that in a few years there will be about 100 commercial nuclear power plants in operation and they all contribute to the data base for their full 40-year design life. This would produce only 4000 reactor years of total amassed unit test time. Perhaps that could be doubled by the world experience, which may or may not be appropriately comparable data. The point estimate for this comparison is only $\lambda = 1/8000 \sim 1 \times 10^{-4}$. The implication, of course, is that low failure rates are difficult to measure without an extremely large number of units on test to accelerate the required large test time. The best result that one can be achieved is a $(1-\alpha)$ upper bound.

D.3.2 Intermediate and Small LOCA Failure Frequencies

Theoretically, it should be possible to construct a plot or table of failure frequency against break size which will range from guillotine large pipe rupture to hairline crack. However, the discussion on large LOCA failure frequency has demonstrated at least one of the practical limitations encountered when attempting to attain these parameters. It may be compounding the uncertainty on failure frequency by trying to examine a partitioned (large, intermediate, and small LOCA) failure frequency when attempting to determine if there is an unaccounted for effect of IGSCC in the failure rate.

Because the absolute value is not of interest for the failure frequency for any failure size, but only the relative comparison of the causes contributing to the failure frequency, it is more important to examine the data base rather than the partitioned failure frequency. The examination of the partitioned failure frequency is done thoroughly in a recent Canadian Study (D.3) which is discussed in the next section.

D.3.3 Pipe Failure Data Base

Since publication of the safety study WASH-1400 in 1975, there have been few systematic reviews or updating of the pipe failure rate data base. The timing and resource limitations of this study did not permit a full and detailed updating of the data base, which was inconsistent with the committee objectives. However, it was possible to review available studies and the U. S. nuclear data base as represented in the LERs available in computerized data banks of the current Sequence Coding and Search (SCSS) data base at ORNL.

The most comprehensive study of the U. S. nuclear pipe failures, as represented by LERs, was done by the Canadians and reported in reference D.3. This represented the experience base in commercial U. S. nuclear power plants through December 31, 1978, for a total of 409 reactor years of operation. Although it was not possible, to extend the in-depth analysis performed by the Canadians, it was possible to review the reported pipe events and their causes between 1978 and November 1983 to see if any trends in causes could be obtained.

Reference D.3 noted the difficulties with analyzing the data base, as did S. Bush in a forthcoming publication (D.5). It must be realized that the data base results from conditions far from controlled experimental practice. The first problem, common to both the event recorders and event analysts alike, is common definitions for "failure," "rupture," and "severance." There are other conditions discussed later which may well make the determination of true failure rate highly uncertain. However, these data (presented in Table D.4) do represent the best available information and, with the limitation clearly in mind, should provide insight into the ratio of causes.

Only two major contributors to failure have been separated from the total: vibration/fatigue and erosion/corrosion. Although there does not appear to be a progressive increase in the category designated erosion/corrosion (an abbreviated category as analyzed in reference D.3) which includes IGSCC, there probably is little statistical significance for the following reasons: a) improved reporting requirements may well have some impact on the results; b) there are a large number of events with causes unknown; and c) in

Table D.4

Fatigue and Corrosion Failures
As a Function of Total Reported Failures
(U. S. Commercial Nuclear Power to Nov. 1983)

	To 12/78	12/78 to 10/81	10/81 to 11/83
Total Events	804	138	98
Vibration/Fatigue	10%	25%	24%
Erosion/Corrosion	13%	15%	22%
Operating History (Rx-yr)	409	540	700

some events the causes are speculative because reporting deadlines do not allow sufficient time for analysis to specify the cause accurately. Failure analysis which lags the LER submission is rarely updated and correlated with a revised LER. In addition, a sizable number of IGSCC events are uniquely associated with the PWR boron injection system. Therefore, there is large uncertainty associated with the pipe break causes. Further, the uncertainty in the failure frequency as inferred from the chi-squared test is very large by comparison with any trend in the erosion/corrosion contribution. To attribute statistical significance to a 25% change in a partitioned component of a failure rate which has two orders of magnitude uncertainty in its value cannot be statistically supported. It should be noted that the removal of <u>incipient</u> failures by UT or other in-service inspection and subsequent repair is not included in these data.

D.3.4 Leak-Before-Break

The small LOCA failure rate represents a difficult problem. In general, defects including IGSCC have a greater likelihood of forming a small opening (crack or hole) and leaking well in advance of a major failure (break). If leak detection systems function properly, this leakage is detected and corrected well in advance of becoming even a minor problem. This is generally referred to as the "leak-before-break" concept and appears to be borne out both in theoretical calculations (reference D.7, p. 196) and in the field data (reference D.3, p. 68). However, the methodology defined by reference D.1, and apparently followed in most other PRAs, defines the BWR small-break LOCA as less than 2-1/2- and greater than 1/2-in. equivalent pipe diameter. Most failures recorded are very difficult to categorize, at least with any precision as to size, because of the reporting system.

The leak-before-break may theoretically precede any size break and the leak itself could be large enough to qualify for a small or intermediate size break (although, this is unlikely assuming the leak detection system is functioning properly), but the quality of the data base is inadequate for one to make such distinctions. Because of the data base quality and the artificial division of break sizes, no conclusions about absolute failure rates will be attempted. However, inferences about relative causes for failures can be estimated even though the data base may be uncertain for absolute failure rates.

D.4 SYSTEMS RISK ASSESSMENT

D.4.1 Plant Model Selection

There are approximately 40+ BWR Plants licensed to operate or in the licensing process. As the BWR population has grown, the reactor manufacturer has introduced improved models in his product line so that there are currently 6 classes of BWRs in the 40+ plant population. In order to draw generic conclusions from a study of this sort within the resource and time limits available, it is necessary to select a characteristic plant for study. Generically, this is not too difficult because the type and number of systems have not changed markedly for the last three models (BWR 4-6) which comprise about three fourths of the population. Roughly half this population are BWR-4s. Therefore, a BWR-4 was selected as a model. This is also consistent because the RSS selected a BWR-4, Peach Bottom-2, for the BWR risk assessment. The methodology for PRA developed in the RSS, at least in the LOCA sequences, has been used in all other plant-specific BWR PRAs reviewed, which include Peach Bottom-2 (RSS), Millstone 1, Browns Ferry, and GESSAR (BWR-6). The plant selected as a model was the Georgia Power Company's Edwin I. Hatch Nuclear Power Plant Unit 2, which went into commercial operation in 1979. The Hatch-2 plant is also significant because Georgia Power Co. has elected to replace their recirculation system with a new system, using nuclear grade 316 stainless steel, which is designed to reduce the number of welds in the system.

From a systems standpoint, the changes which have occurred during the development of the BWR-3 through -6 have caused only nominal impact on the number and type of piping systems significant to safety. The BWR-3 was the first large direct cycle system utilizing jet pumps internal to the reactor vessel and improved ECCS, including core spray and flood capability. The BWR-4 was much like the BWR-3 with increased power density. The BWR-5 and -6 product lines incorporated flow control, changes in fuel design, and changes in pressure suppression/containment design. The BWR-4 also incorporates the major primary engineered safety and auxiliary systems followed in later designs. This, in conjunction with the fact that more BWR-4s are operating than other models, makes the selection of a BWR-4 the logical choice.

D.4.2 Previous Pipe Crack Study Groups

The efforts of previous pipe crack study groups (PCSGs) were reviewed. Each PCSG emphasized certain areas according to its charter, the field experience to that point, and the resources marshalled for the study scope. In some cases, pipe cracking in both PWR and BWR reactor types (D.8) was considered and in others only one reactor type (D.9 and D.10). The most recent PCSG focused on the cracking implications in PWRs (D.10). In its study (D.10), the PCSG placed major emphasis on systems safety implications of cracking, essentially auditing the deterministic licensing methods in use at that time. The systems approach for the current PCSG centers on the IGSCC implications in BWRs and its risk implications. Therefore, the systems assessments are made in the framework of the RSS risk methodology. This reflects the trend of assimilating risk methods into the licensing process. The RSS methodology, in this context, has some powerful attributes and it also has some deficiencies. Examples of both strength and weakness will illustrate this point.

Risk methods allow an evaluation beyond the deterministic concepts of single failure and redundancy. The risk methodology allows not only a determination of significance to safety, but also quantifies what is meant by the statement. On the other hand, risk methodology is statistical in nature and is applied in the aggregate to a plant. This becomes very difficult when trying to partition the risk to the subsystem level such as pipe failure frequencies. In its global concept, it also yields such things as what action(s) are necessary to retain that importance to safety throughout the life of the plant with respect to test, maintenance, and operator actions in emergencies.

When the RSS was published, the peer review produced much comment about the intrinsic large uncertainties associated with the methodology and the unreliability of the absolute values. These are two well-publicized difficulties with risk assessment techniques. In applying these techniques to the cracking of pipes, it was difficult to partition the risks to both phenomenon and subsystem. For example, the failure frequencies for pipes normally include all causes: fatigue, corrosion, erosion, design defect, etc. Deterministically, existing procedures and approaches are intended to reduce the causes of failure to an acceptably low level. Statistically, however, it is necessary to quantify these very infrequent events. In order to gather a data base on these failure frequencies with the lowest uncertainty possible, the events are aggregated with no attempt to develop failure frequencies for each cause.

Finally, the LOCA-sensitive piping is lumped together in only two categories, large and small, and not by subsystem. This is primarily for convenience; but to make relative comparisons, the studies made under existing methodology must be used. As PRA becomes a more acceptable tool, a consistent LOCA initiation probability by subsystem piping within the RCS may be made.

The following discussions of the systems implications of risk assessment will rest heavily on the Hatch-2 plant. It should be kept in mind that these discussions cannot be taken in isolation from other sections of this report and that certain significant plant specific deviations from these discussions may exist.

D.4.3 Systems Evaluations

Up to this point, attention has been directed primarily to the initiating event in the reactor accident sequence. In the cases of IGSCC (or other contributors), the pipe failure or rupture is considered to be the initiating event without regard to precursor mechanism. This has been common PRA methodology from the RSS through current risk assessments reviewed. It is not within the scope of this study to argue either for or against the validity of this initial condition, but for most conditions pipe failure is initiated by the imposition of some pressure, temperature, or mechanical load which, in turn, will cause the defective pipe to fail. This approach implies that all initiating causes are included when the consequences of pipe failure are calculated using the piping failure frequency. The pipe failure leads to a loss of coolant. If the failure of the pipe has certain characteristics,* a loss-of-coolant accident (LOCA) occurs.

The systems of concern for a BWR are listed in Table D. 5, which is extracted from NUREG-0531 (D.9) and pertains to a large BWR-4 as noted earlier.

The most severe LOCA, in terms of peak clad temperature (when the ECSS works) is the DBA. The BWR DBA is a double-ended guillotine (DEG) severance of the suction line of B main recirculation pump (D.1). The BWR is designed to be able to sustain the DBA without unacceptable damage to the reactor fuel and bring the plant to a cold shutdown by use of the ECCS and long-term heat removal capabilities. The ECCS is an integrated group of systems described in the FSAR of all plants and is designed to provide coolant makeup for a variety of LOCA conditions up to and including the DBA. The systems which comprise the ECCS include the following: High Pressure Coolant Injection (HPCI), Automatic Depressurization System (ADS), Core Spray (CS), and Low Pressure Coolant Injection (LPCI). Briefly, each system will be discussed to illustrate how these systems provide adequate coolant makeup for any LOCA including the DBA. In order to focus the reader's attention, Figure D.1 is a composite plot which shows the LOCA size, flows, and leak detection limits as taken from the appropriate documentation (D.11). The following LOCA descriptions pertain to Figure D.1.

*For generic discussion, it is not necessary to stipulate what all these characteristics are; for example, that if the leak is large enough, it may go undetected, etc.

Table D.5

List of Typical Piping Systems Involved in the Operation and/or Safe Shutdown of BWRs*

Identification Number	Function
1	Reactor Recirculation
2	Main Steam
3	Feedwater
4	Reactor Water Cleanup
5	Reactor Core Isolation Cooling
6	Core Spray
7	Residual Heat Removal
8	Containment Spray
9	Reactor Head Spray
10	Standby Liquid Control
11	High Pressure Coolant Injection
12	Low Pressure Coolant Injection

*This list is taken from the FSAR of Peach Bottom Units 2 and 3.

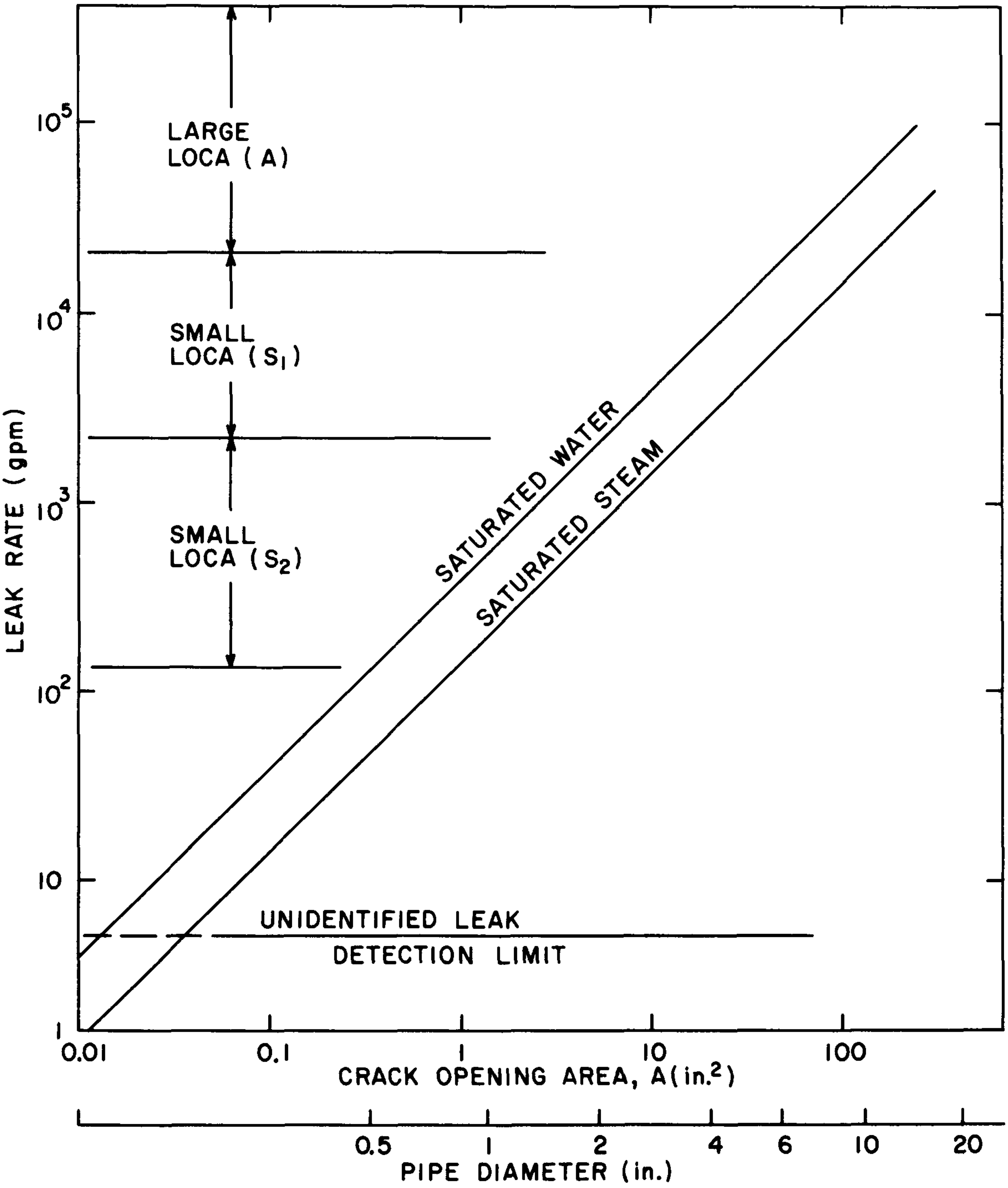


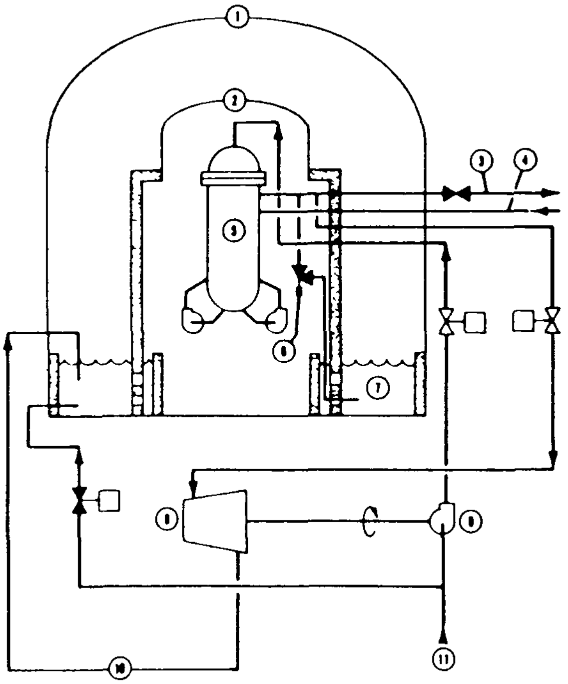
Figure D.1 Leak rate as a function of crack opening area or pipe diameter (assuming a full circumferential break).

For small-small LOCA (S_2), the coolant loss can be made up by non-ECC systems of feedwater and/or control rod drive cooling water until the plant can be shut down and the leak located and repaired.

For small LOCA (S_1), the HPCI system will be activated to supply coolant until the plant can be secured and the leak found and repaired.

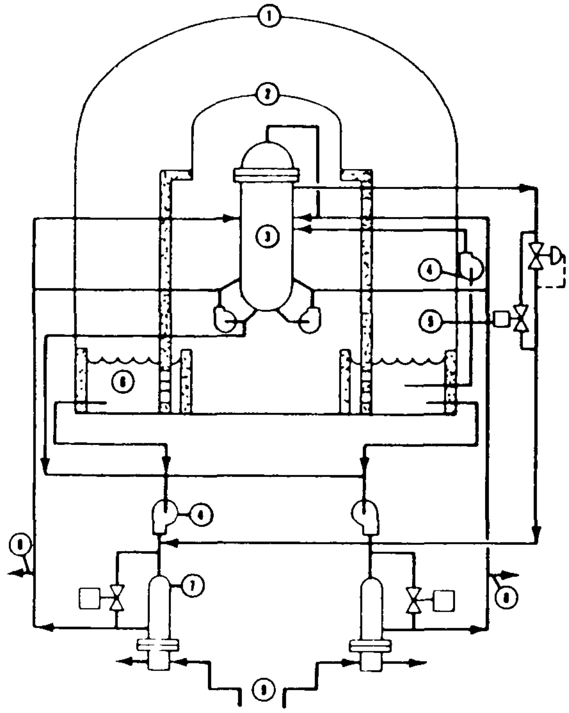
A large LOCA (A) is of sufficient magnitude that adequate coolant level cannot be maintained (automatically sensed and initiated by low reactor vessel water level or high drywell pressure), and the ADS reduces the reactor vessel pressure through selected safety relief valves so that coolant can be supplied by both the LPCI and/or core spray.

The ECCS is designed to cool the reactor core for LOCAs which range over the entire spectrum up to and including the DBA. After the initial phases of the LOCA (as well as many other events), the long-term cooling is accomplished by the Residual Heat Removal (RHR) system. The RHR system is a multi-functional system which includes the LPCI mode of operation. Briefly, the other RHR functional modes are: 1) steam condensing and 2) suppression pool cooling, in addition to the shutdown cooling and LPCI functions already noted. The RHR systems functions are shown in Figures D.2 through D.7 for a BWR-6, Mark III containment. For the purpose of this discussion, there is no functional difference compared with a BWR-4, Mark II containment. The steam condensing function is unimportant to this discussion because the possible IGSCC-LOCA concern deals with systems using 304 stainless steel. The steam piping in all plants reviewed is low-carbon steel. It is of interest to note,



1 CONTAINMENT
2 DRYWELL
3 MAIN STEAM
4 FEEDWATER
5 RPV
6 SAFETY/RELIEF
7 SUPPRESSION POOL
8 TURBINE
9 TURBINE DRIVEN MAKEUP PUMP
10 TURBINE EXHAUST
11 CONDENSATE FROM CONDENSATE STORAGE TANK OR RHR HEAT EXCHANGER

Figure D.2. Reactor core isolation cooling system.



1 CONTAINMENT
2 DRYWELL
3 RPV
4 SYSTEM PUMPS
5 FULL FLOW BYPASS
6 SUPPRESSION POOL
7 HEAT EXCHANGERS
8 RCIC PUMP SUCTION
9 SERVICE WATER

Figure D.3. Residual heat removal system.

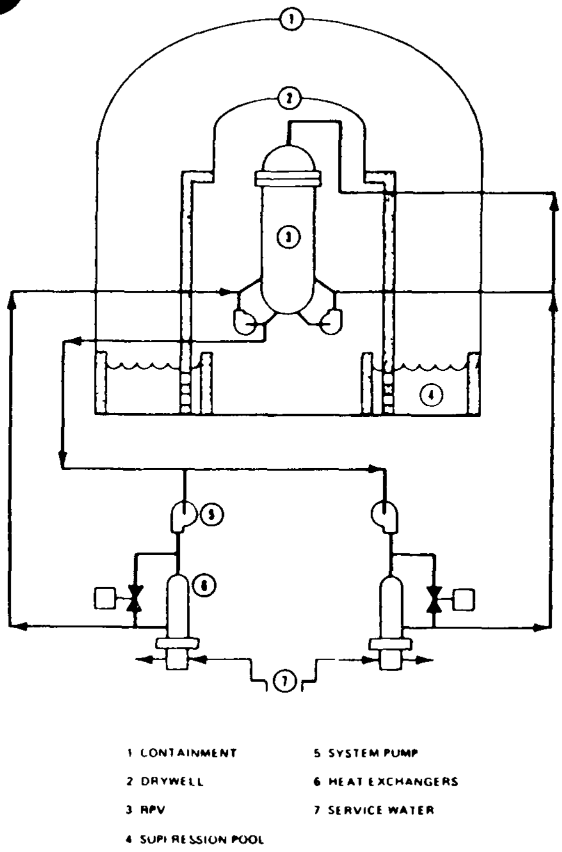


Figure D.4. Residual heat removal system, shutdown cooling function.

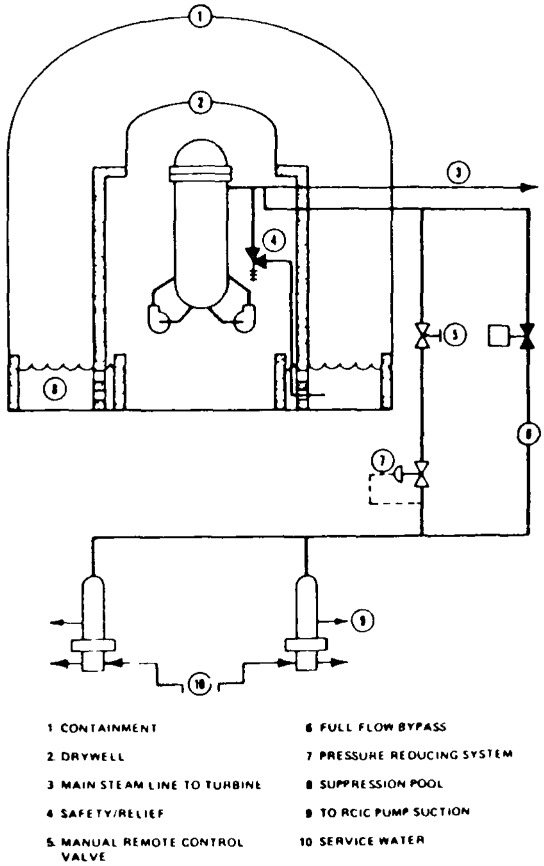


Figure D.5. Residual heat removal system, steam condensing function.

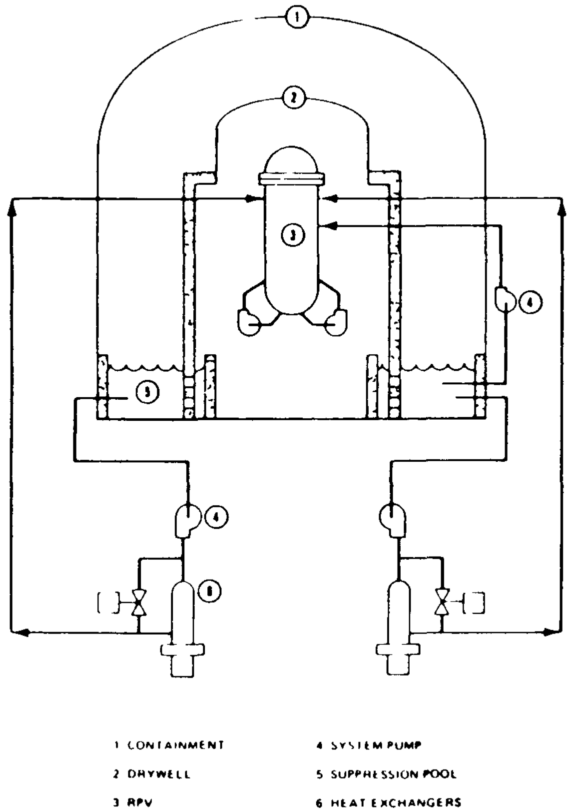


Figure D.6. Residual heat removal system, Pressure coolant injection function

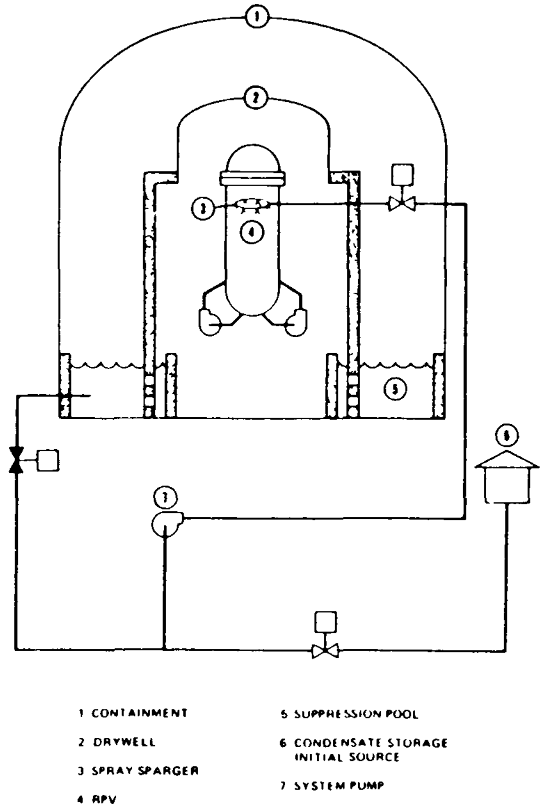


Figure D.7. High pressure core spray system.

however, that a recent study (D.12) which sampled a five-year window from 1976 through 1980 for loss-of-coolant events found that these events were dominated by the main stream relief valves. These events constitute a subclass of LOCA initiators which are included in the failure frequencies by their valve seat area equivalent pipe diameter. These are generally in the RSS category of "small" breaks (between 1 and 8-1/2-in. in diameter).

For the sake of completeness, we include another system associated with the BWR called the Reactor Core Isolation Cooling (RCIC) system (see Figure D.2). The RCIC system maintains sufficient water in the reactor pressure vessel to cool the core and maintain the nuclear boiler in the standby condition in the event the vessel becomes isolated from the turbine steam condenser and from the feed-water make-up flow. The RCIC operates in conjunction with the steam condensing mode of the RHR. It is not necessary to consider these modes of functioning of the RHR further because they are unimportant to the LOCA sequence.

Reviewing, first, the field experience of IGSCC, some generalizations are possible. IGSCC has been found, primarily, in the two recirculation loops. Until 1978, IGSCC had not been discovered in large-diameter piping. Since that date, IGSCC has been found in the large-diameter piping in numerous plants. IGSCC appears to be more prevalent in small-diameter piping. However, there appears to be considerable variation in this generality due, possibly, to the number of variables affecting the susceptibility and progress of the IGSCC phenomenon.

Further, no IGSCC has proceeded to a size sufficient to be classified as a loss-of-coolant event which would activate the ECCS. All known events were detected by either UT inspection or very small leakages ($\leq$ 10 gpm) (D.13). Total avoidance of IGSCC attack is preferable; however, experience would suggest strongly that both leak detection (or conversely, leak-before-break) and inservice inspection are effective as components of the defense-in-depth concept. The thrust of this discussion is neither leak-before-break nor inservice inspection, both of which are discussed in Sections 6 and 4, respectively, but to analyze the systems implications of IGSCC. A review of reference D.13 and subsequent IGSCC detections suggests that the attack is centered in the recirculation and associated systems piping with no involvement in the important safety systems in Table D.5, with the exception of the core spray system. The reactor water cleanup system has been found to be heavily involved (D.9), but is not considered a primary safety system.

From a risk standpoint, a crack in a pipe can, basically, contribute to risk in one of two ways. The crack can be a LOCA initiator or it can disable an important safety system.* The contribution to the LOCA frequency is adequately addressed through the data base and application to the LOCA sensitive piping as outlined in the RSS (D.1). IGSCC has been postulated as a phenomenon which could produce multiple LOCA events by the Advisory Committee on Reactor Safeguards (ACRS) (D.14) and others. Multiple pipe failures as a result of IGSCC warrant some discussion and will be addressed later.

* Secondary concerns associated with systems interactions and common mode failure are not being considered here.

The probability of safety system failure (unavailability) is an integral part of risk assessment. The initial and current licensing philosophy is based on what is usually termed a deterministic approach. Briefly, it relies on a thorough mechanistic analysis and evaluation of the nuclear power plant, as well as the application of the concepts of system redundancy, diversity, and independence to systems important to safety to ensure their performance. These systems and subsystems must be capable of performing their functions even for the most severe single-component failure. These deterministic considerations define an envelope termed the DBA spectrum. Engineering judgment and conservatism are used in determining whether the plant design is acceptable in meeting the general design criteria for LWRs as published in 10 CFR 50, Appendix A. Risk assessment quantifies the probability that severe accidents might occur and what their consequences would be to the public. A PRA attempts to analyze and quantify all aspects of the nuclear plant design, operation, and maintenance for internal (and external) initiators of severe accidents which would result in residual risk to the public.

PRA techniques have provided numerous useful applications (D.4), including methods which allow a quantitative measure of the system importance to safety (risk). Table D.6, derived from PRA allied techniques (D.15), provides a qualitative ranking of the LOCA-related ECC systems. The reactor protection system has been included for comparison, since it is without question the most important safety system in the plant as demonstrated by this table. Risk achievement ratio is defined as the factor by which core-melt frequency would increase if the system were never operable. Although this ranking is for a BWR-6 which might alter the Risk Achievement Ratio values, the relative ranking for Hatch-2 (the selected model) would not be expected to be significantly different.

Table D.6

Risk Worth for Safety Systems* with Respect to Core Melt Frequency (D.15)

System	Risk Achievement Ratio
Reactor Protection System	2.0×10^5
Residual Heat Removal System (RHR)	1500
Standby Service Water System	1500
Low Pressure Coolant Injection (LPCI) System	23
Automatic Depressurization System (ADS)	14
Low Pressure Core Spray (LPCS) System	7.6
High Pressure Core Spray (HPCS) System	4.4
Reactor Core Isolation Cooling (RCIC) System	2.2

*Based on Grand Gulf, a BWR-6, which has HPCS rather than HPCI as in BWR-4 as well as some minor differences, but safety systems/functions are similar. Some systems not directly pertinent to LOCA concerns have been omitted from this list.

Table D.6 in conjunction with the field data discussed earlier, can be used to obtain some measure of the systems impact of IGSCC. IGSCC has been found in the core spray systems of some plants. Although IGSCC in the core spray systems should not be ignored, its presence does not increase the risk by a substantial amount because: 1) the system is redundant, 2) core flooding may be accomplished by other systems, and 3) a crack in these systems (assuming leak-before-break) would most likely only degrade the system performance.

Pipe cracking or failure in a system is generally less severe than the failure of valves to operate. The ECCS must be capable of performing even with single valve failures. Table D.7 presents the results of the single-valve-failure analysis for Hatch-2 (D.16).

On the other hand, although the RHR system has experienced little IGSCC attack, it is extremely important as noted by its risk achievement ratio and should continue to be inspected for IGSCC or other faults. The Reactor Water Cleanup (RWCU) system has had a considerable number of IGSCC events. That part of the RWCU system that cannot be isolated from the recirculation system has been included in the LOCA-sensitive piping. The RWCU system contributes only nominally to risk in any other respect.

Table D.7

ECCS Single-Valve-Failure Analysis

System	Valve(s)	Position for Normal Plant Operation: Closed	Position for Normal Plant Operation: Opened	Consequences of Valve Failure Assumed Together with Design Basis LOCA
Core Spray	Suction		X	Negate use of the core spray loop
	Injection(s)	X		Negate use of one core spray loop
	Test return	X		Negate use of one core spray loop
High pressure coolant injection	Condensate suction		X	Use suppression pool water
	Suppression pool suction valve	X		Use condensate storage tank water
	Suppression pool test return	X		Partial flow loss due to flow to suppression pool
	Injection(s)	X	X	Negate HPCI
	Turbine inlet(s)	X	X	Negate HPCI
Low pressure coolant injection	Injection(s)	X		Negate use of LPCI
	Minimum flow		X	Partial flow loss in one loop due to flow to suppression pool
	Test return	X		No consequences
	Pump suction		X	Negate one out of four pumps
Automatic depressurization system system	One relief valve	X		Vessel depressurizes faster

D.4.4 Multiple Pipe Failures

The concern with multiple pipe failures stems principally from the possibility that IGSCC damage could be present in more than one location and in more than one piping system. If the plant were to experience a severe loading event, such as an overpressure transient or an earthquake, the plant loading event might lead to leaks or breaks at more than one location at essentially the same time.

A reduction of safety margins of this type is clearly undesirable if it can be prevented. However, from a risk standpoint and for this discussion, it must be assumed that the possibilty exists to some degree. The following will consider that likelihood. The chief concern may be limited to two possibilities: a) that a LOCA larger than the DBA results from a multiple failure or b) that a multiple failure causes not only a LOCA, but the second break disables an ECC system.

The concurrent LOCA DBA and the disabling of an ECC system is very similar to the single failure criteria already required of the ECCS discussed earlier. As pointed out in that discussion, this induced pipe failure is no more severe than a valve or pump failure, conditions already included in both the deterministic and PRA assessments. Gerber and Garud (D.11) point out that ECCS redundancy leaves adequate system coverage for this concern and discuss the probability of multiple pipe failures. They note that considerable margin exists both in ECCS design and conservatisms in peak clad temperature calculational methods.

With respect to the probability of multiple failures, Gerber and Garud present a statistical argument which suggests that the probability of two concurrent failures lies bounded at most by the probability of the least probable single failure and is at least equal to the product of both single failure probabilities. Reference D.11 includes a lengthy discussion of the statistical dependence and the correlations of variables involved which will not be repeated here.

As an example of estimating the probability of two concurrent pipe failures, assume that each break has the same single failure rate distribution [i.e. $P(f_1) = P(f_2) = P(f)$]. With this assumption, the joint failure probability can be reasonably approximated as a lognormal distribution. Utilizing the statistical methods and approach of the RSS (D.1) and the above Gerber-Garud statistical argument regarding statistical correlation, the median of the joint probability for two concurrent failures can be expressed as

$$P(f_1,f_2) = \sqrt{UL} = \sqrt{P(f) \times P^2(f)} = \sqrt{F_L \lambda_{LP}} \times P(f) , \qquad (11)$$

where f_1 and f_2 represent failures 1 and 2, U is the upper bound [$P(f)$] and L is the lower bound [$P^2(f)$], F_L is the fraction of large pipe which is LOCA sensitive, and λ_{LP} is the large pipe failure rate. The values used in the RSS are $F_L = 0.047$ and $\lambda_{LP} = 1 \times 10^{-4}$. Substituting these numbers into (11) results in

$$P(f_1,f_2) \cong 2 \times 10^{-3} P(f) , \qquad (12)$$

which indicates that the contribution to the core melt frequency for concurrent or multiple large pipe ruptures is about 3 orders of magnitude less than that associated with a single large pipe rupture.

Although these arguments are approximate in nature, they indicate that the risk contribution of multiple large failures associated with IGSCC is very likely several orders of magnitude less than that of a single large failure. Approximate though these arguments may be, they are probably much better than would be obtained if one tried to extract a joint probability from the current or foreseeable data base. However, there are statistical codes on reliability, such as PRAISE (D.7), with which this approximation may be concurred.

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APPENDIX E

VALUE-IMPACT ASSESSMENT

This appendix describes each postulated IGSCC mitigation scenario and associated assumptions. It presents the unit ORE and dollar costs for the various inspection and mitigation procedures considered. It also discusses the analysis of each factor analyzed.

E.1 IGSCC RESOLUTION SCENARIOS

The general assumptions as well as the activity sequences and assumptions associated with each postulated scenario are described below.

E.1.1 General Assumptions

Several assumptions related to plant design and operations are needed to complete the analysis. For this purpose, it was assumed that BWR plants are shut down for six weeks to refuel once every 18 months and that the plants undertake every third refueling cycle a major maintenance activity such as turbine-generator inspection/maintenance or main condenser tube replacement which lasts three months. It was further assumed that the average remaining life of operating plants is 25 years (E.1). The assumed average number of welds in sensitive material in a typical operating BWR is shown in Table E.1.

Table E.1

Average Operating BWR Plant - 304 SS Piping Welds $\geq$ 4 in.*

System	Total Welds, Av.	No./size	No./size
Recirculation	98	42-(22-28 in.)	56-(>4-12 in.)
Recirculation Bypass	8	8 @ 4 in.	
Core Spray	9	9 @ 12 in.	
RHR	25	10-(20-24 in.)	15-(6-20 in.)
RWCU	20	19-(4-10 in.)	
	160		

*Average number per plant and size of 304 SS pipe welds (4 in. and up) derived by averaging information from IE Bulletins 82-03/83-02 inspection reports and from GECo for 22 operating BWRs.

E.1.2 Scenario 1 - Pipe Replacement

1. Operating Sequence. This scenario assumes that all sensitive piping is replaced. HWC was considered as an option. Pipe replacement was assumed to require the following activities.

a. Decontaminate and remove old pipe and replace with less sensitive material.

b. Stress-relieve field welds using IHSI process or the equivalent treatment.

c. Perform UT examination to establish a baseline for subsequent inspections.

d. Resume operation.

2. Plant Outage. Total plant outage for replacing all 304 SS piping (greater than 4 in.) inside the primary pressure boundary is estimated to last 6-9 months. Pipe replacement is assumed to be scheduled concurrent with a refueling outage of 6 weeks or with other major maintenance outage such as turbine-generator inspection/maintenance or main condenser retubing lasting 3 months. Thus, the incremental plant outage time chargeable to pipe replacement was assumed to range from 3 to 7.5 months with a best estimate of 4.5 months. Cost for replaced power is assumed to be $300K per day.

3. Reduction of Pipe Welds. As a result of pipe replacement, the average number of pipe welds inside the primary pressure boundary is assumed to be reduced from 160 to about 120 welds per plant subject to post-replacement ISI. The basis for this assumption is:

a) As part of pipe replacement, plants plan to eliminate the Recirculation system discharge valve bypass piping which eliminates 8- to 4-in. pipe welds.

b) Use of design improvements in replacement pipe fittings will eliminate 30-35 (assume 32) recirculation system pipe welds of which 20 are 12 in. in diameter and the remainder are 22 to 28 in. in diameter. The improvements include features such as use of bent 12-in. pipe instead of welded elbows; elimination of 22-in. welded end caps on riser headers; use of integral long tangent 28-in. elbows instead of welded tangents.

4. Post-Replacement Inservice Inspection. Plants that replace 304 SS pipe with acceptable IGSCC-resistant pipe material were assumed to be allowed to reduce their inservice inspection (ISI) to normal ASME Section XI requirements. The Code provides for inspection of 25% of pipe welds over each 10-year period of plant operation. On an 18-month refueling cycle, there would be 6.67 refueling cycles each 10 years, which implies an average inspection rate of 3.75% of the welds each refueling. However, the Code does not require that inspections be performed each outage. For this pipe replacement scenario, it was assumed that at least 4% of the welds would be inspected in each of the first two outages. The frequency of inspection was then assumed to follow the Code requirements for the remainder of the plant life. Inspections were assumed to be performed during a scheduled outage at least once every 3-1/3 years and cover 8% to 9% of the welds in order to meet the Code requirements. This level of inspection was not assumed to extend these outages. It was assumed the pipe replacement would reduce subsequent defects and repairs to negligible levels.

5. Hydrogen Water Chemistry (HWC) Option. An option in the pipe replacement scenario is the addition of HWC. Operation of this system with replaced piping is expected to provide a small additional benefit in the reduction of piping degradation. However, it is believed to warrant consideration for avoidance of potential long-term stress corrosion of core internals and would provide an additional degree of certainty in avoiding piping problems. For this reason, it was assumed that if a plant added HWC

that ISI requirements would be reduced to ASME Section XI immediately following piping replacement. Like pipe replacement alone, this option was assumed to reduce new defects and repair requirements to negligible levels. The effect of increased main steam gamma activity due to increased 16N resulting from HWC operation is discussed in Section E.3 below.

E.1.3 Scenario 2 - IHSI With HWC

1. Operation Sequence. This scenario assumes that all sensitive welds have been previously inspected and found to be without defects. The welds are treated with IHSI or equivalent techniques to desensitize them to IGSCC damage. The following activities are assumed to be performed as part of this scenario:

   a. Decontaminate the pipe as an optional but ALARA-recommended measure prior to IHSI.

   b. Treat all welds using IHSI or an equivalent process.

   c. Inspect the piping to establish a baseline UT record for future inspections.

   d. Install and operate HWC system.

   e. Resume operation.

2. <u>Plant Outage</u>. The total outage for this scenario was assumed to last 3 to 5 months. Like Scenario 1, this activity could be conducted concurrently with major maintenance or during a normal refueling outage. This results in an assumed outage chargeable to IHSI and HWC implementation of 0 to 3.5 months with a best estimate of 2.5 months.

3. <u>Post-IHSI ISI</u>. ISI requirements following implementation of this scenario are assumed equivalent to the case of pipe replacement without HWC. Assumed inspection requirements would include 4% of the welds for the first two refueling outages followed by reduction to the ASME Section XI schedule. ISI work is assumed to be conducted every other outage. It was assumed that subsequent defects and required repairs would be reduced to negligible levels following this treatment.

4. <u>IHSI Without Hydrogen Water Chemistry</u>. An option in this scenario is not to use HWC. This was assumed to have a negligible effect on the outage time for implementation. However, the post-IHSI ISI requirements are revised to assume inspection of a minimum of 8% (double the ASME XI average) of the welds for the first four refueling cycles to obtain field data to confirm the long term effectiveness of IHSI alone. Subsequent inspections would reduce to the normal ASME XI scope and schedule. For these inspections, it was assumed that one new defect would be found for every 20 welds inspected and that repairs would be needed for two out of three defects found. The assumed defect rate of 1/20 is about one fourth that experienced through December 1983 by plants that performed IE Bulletins 82-03/83-02 inspections. The assumed repair rate of two-thirds repair per defect is the same as IE Bulletin

inspection experience through December 1983. No incremental outage time has been assumed for post-IHSI inspections; however, 0 to 2 weeks with a best estimate of 1 week was added to the outage when repairs were required.

E.1.4 Scenario 3 - Augmented Inspection and Repair without Hydrogen Water Chemistry

1. Operations Sequence. In this scenario current inspection and repair practices are continued in order to mitigate the effects of IGSCC. This is viewed primarily as a short-term option while a decision is being made between the previous long-term options. It is mainly based on the assumption that an inspection method or basis for accepting weld overlay repairs for more than one refueling outage will be established. The following activities are assumed to be completed as part of the scenario:

   a. Continue inspections similar to IE Bulletins 82-03 and 83-02.

   b. If defects are discovered, evaluate the possibility of propagation using fracture mechanics analysis. If repair is not required, record for future monitoring.

   c. If repairs are required, use the weld overlay technique or equivalent. Repaired welds would be inspected before resuming operations.

2. Inspection Scope and Frequency. From the following, it is assumed that 50% of the average number of welds (160) of a typical BWR will be examined by UT during each refueling outage:

It is assumed that the augmented ISI plan presented in Appendix C will be applied at each refueling outage for plants that do not implement measures for long term mitigation of IGSCC. This augmented ISI plan basically requires inspection of an initial 20% sample of welds and previously uninspected and uncracked welds plus inspection of unrepaired cracks and all weld overlay repairs on cracks measured to exceed 10% of the pipe circumference. Further, if new cracks or significant growth of old cracks is found in the initial 20% sample, the sample size should be expanded in accordance with IWB 2430 of ASME Code Section XI. Experience from IE Bulletin 83-02 which specified an initial sample of 22 welds indicates that on average the plants actually inspected four times the number of the initial sample size. However, it is expected that the extent of sample expansion for future inspections will be reduced. This expectation is based on the fact that the IE Bulletins 82-03/83-02 inspections examined a large number of welds for the first time using a UT technique that is much more sensitive in detecting IGSCC than any technique used previously. Also, some licensees decided to inspect 100% of the recirculation/RHR system welds regardless of the IE Bulletin specified sample size. Thus, for this scenario it is estimated that future inspections will examine on average twice the initial 20% sample plus the previous repaired and unrepaired cracks. This could amount to about 50% the average number of welds, for a typical plant or about 80 welds per plant during each refueling outage. The inspection scope was assumed to range from 40% to 60%.

3. Defect Rate. Experience through December 1983 from IE Bulletins 82-03/83-02 inspections indicated an average defect rate of 19% (283 defects for 1509 welds inspected). It is assumed that future inspection will average a reduced defect rate of about 10% because of the high level of inspection completed under the Bulletins and the fairly short time (18 months) between successive inspections.

4. Repair Rate. The repair rate experienced under the Bulletins is assumed to remain unchanged at about two repairs every three defects (183 repairs per 283 defects).

5. Plant Outage. Outage time for this scenario depends on the number of defects that are discovered and repaired. This level of inspection and repair was assumed to result in three to four months of total outage. The outage time chargeable to inspection/repair depends on whether the work is done concurrently with a refueling outage (six weeks) or a major plant maintenance activity (three months). It was assumed that major maintenance occurs every third refueling cycle. Thus, the assumed incremental outage time, in weeks, is as follows:

	First Cycle	Second Cycle	Third Cycle
Lower Estimate	6	6	0
Nominal Estimate	8	8	2
Upper Estimate	10	10	4

6. <u>Hydrogen Water Chemistry (HWC) Option</u>. An option in the inspect/repair scenario is to install and operate an HWC system. HWC installation is not anticipated to result in additional outage or significant ORE. ISI frequency would remain as indicated above. However, it is expected that HWC will result in the presence of fewer defects at each inspection. Thus it was assumed that the defect rate would reduce to one new defect for every 20 uncracked welds inspected. This should eliminate or greatly reduce expansion of the original 20% sample size. Thus, it is assumed that on on average 30% of the welds are inspected: the initial 20% sample <u>plus</u> 10% to reexamine previous indications and repairs. The inspection scope was assumed to range from 20% to 40%. The repair rate of two repairs for every three defects was maintained. It was assumed that the outage time necessary to complete these inspections would reduce each refueling cycle two to three months. Using the same assumption as above for other major maintenance activity in the plant, the incremental outage, in weeks, is as follows:

	First Cycle	Second Cycle	Third Cycle
Lower Estimate	2	2	0
Nominal Estimate	4	4	10
Upper Estimate	6	6	0

E.2 UNIT VALUES FOR ORE AND INDUSTRY COST

This section develops unit value estimates for ORE and costs for weld inspection, weld repair, pipe replacement, IHSI, and HWC. The unit values as well as estimated plant outage times are based on reports and verbal communication from licensees, EPRI, the NSS vendor, and some subcontractors engaged in various activities resulting from IE Bulletins 82-03/83-02. They also utilize estimates of utilities planning to replace IGSCC-sensitive pipe. These results are used in the following sections to estimate cost impacts of scenarios in this value-impact assessment.

E.2.1 UT Inspection

The UT experience of several plants is summarized in Table E.2. The ORE data are based primarily on an EPRI tabulation of UT inspection experience for 15 plants. The cost data are based on approximate cost information provided by three plants.

E.2.2 Weld Overlay

The weld overlay experience is summarized in Table E.3. The ORE data are based primarily on an EPRI tabulation of weld overlay experience for 15 plants. The cost data are based on approximate cost information provided by two plants.

Table E.2

UT Inspection Summary Data

Plant	ORE Total man-rem	No. of Welds UT'd	Man-rem Weld	Total Cost, $K	UT Cost Weld, $K	Decon
15 plants(1)	1011	1312	0.77	--	--	--
Plant A	73(2)	101	0.72	225	2.2	no
Plant B	100(2)	216	0.46	501	2.3	no
Plant C	248(2)	82	3.0	470	5.7	- not initially - yes subsequently

	Average	Low	High
ORE - $\frac{\text{Man-rem}}{\text{weld}}$	0.8	0.3(3)	3.0(3)
Cost $\frac{\$K}{\text{weld}}$	3.0	2.2	5.7

$$\text{Average defect rate}^{(4)} = \frac{\text{283 defects}}{\text{1509 welds inspected}} = 0.19.$$

(1) From EPRI summary of 1/12/84, revised to reflect specific plant updated data.
(2) Included in EPRI summary.
(3) High-low within 15 plant data.
(4) From IE Bulletin 82-03/83-02 inspections through December 1983.

Table E.3

Weld Overlay Summary Data

Plant	ORE Total	No. of Welds Overlayed	Man-rem Overlay	Total Cost, $K	Cost Overlay $K	Decon
15 plants(1)	2061	200	10.3	--	--	--
Plant B	330(2)	54	6.1	4100	76	no
Plant C	185(2)	22	8.4	1700	77	yes

	Average	Low	High
ORE - $\frac{\text{Man-rem}}{\text{weld}}$	10.3	2.4(3)	19.5(3)
Cost $\frac{\$K}{\text{weld}}$	76	76	77

$$\text{Average defect rate}^{(4)} = \frac{183 \text{ repairs}}{283 \text{ defect}} = 0.65.$$

(1) From EPRI summary of 1/12/84, revised to reflect specific plant updated data.
(2) Included in EPRI summary.
(3) High-low within 15 plant data.
(4) From IE Bulletin 82-03/83-02 inspections through December 1983.

E.2.3 Pipe Replacement

Pipe replacement ORE and cost estimates were obtained from summaries of NRC meetings with utilities planning to replace pipe and follow-up telephone contact with those utilities. The results are shown in Table E.4.

Table E.4

Pipe Replacement Summary Data

Plant	Estimate Ore for Pipe Replacement Man-rem	Cost, $M	Comments
Plant D	1850	30-35	Replacement scope may be less than Plant E & F
Plant E	1500 - 1750	57	
Plant F	# 2000	40	

	Average	Low	High
ORE, man-rem	1750	1500	2000
Cost, $M	44	35	67

Estimated Outage Time - 6 to 9 months.

E.2.4 Induction Heating Stress Improvement (IHSI)

IHSI ORE and cost estimates were obtained by telephone contact with utilities that have used this technique in their plant. The results are shown in Table E.5 below:

Table E.5

IHSI Summary Data

Plant	ORE Total Man-rem	No. of Welds IHSI'd	Weld IHSI Man-rem	Total Cost, $K	IHSI Cost Weld, $K	Decon
Plant H	240	87	2.8	1600	18.4	yes
Plant G	420	91	4.6	6200	68.1	no

	Average	Low	High
ORE Man-rem / Weld IHSI	3.7	2.8	4.6
Cost, $K / Weld	43.8	18.4	68.1

E.2.5 Hydrogen Water Chemistry (HWC) Option

The estimated costs to install and operate an HWC system were obtained primarily from meetings with General Electric Co. and are summarized in Table E.6 below:

Initial installation costs range from $1 to 3 million depending on the type of system installed to generate the hydrogen injected into the reactor feed water, with an average installation cost of $2M.

Similarly, annual operating costs range from $100K to $500K per year with an assumed average of $300K per year depending on the type of hydrogen generating system.

Installation and operating costs are combined as follows:

Table E.6

HWC Summary Data

	Average	Low	High
Installation Cost, $M	2	1	3
Operating Cost, $K	300	500	100

It is assumed that HWC installation will incur 20 man-rem. ORE effects of HWC operation are discussed later in Section E.3 below.

E.3 DISCUSSION OF VALUE-IMPACT ASSESSMENT

This section develops the quantitative results of the value-impact assessment. The strategy used in the development of low, best, and high estimates was to assume that all parameters were at their lowest, highest, or best levels. This assumption is not particularly realistic but does maximize the range that is covered by the analysis.

E.3.1 IGSCC Public Health Impacts

Public health impacts from IGSCC may occur if loss-of-coolant-accident (LOCA) frequencies are increased. A rigorous understanding of the impacts of IGSCC on LOCA frequency is not known. It is currently the subject of an NRC investigation being conducted by Lawrence Livermore National Laboratory. Until results of this study are available, it was decided to use results of sensitivity studies to bound the impacts of piping failures. Two studies have addressed this question (E.2, E.3).

A comparison of the studies indicated that the results are similar in scope and magnitude. Both studies cover results of risk assessments for Limerick, Shoreham, Millstone 1, Browns Ferry, and Peach Bottom (RSS). In addition, Gaertner (E.2) includes Grand Gulf and estimated that the contribution to LOCA frequencies from IGSCC could be up to 30% based on a review of existing data. Section 7 of this report indicates that the contribution may range from 30%-50%. Sensitivity studies are presented that postulate LOCA frequency increases of up to 10 times the values used in the published risk assessments. This corresponds to IGSCC-induced failure rates 20 to 33 times those supported by historical data. Gaertner also reports a case with LOCA frequencies three times the published values as an indication of the effect of IGSCC failures at 10 times their historical rates.

For the purpose of this study, it was assumed that increasing the LOCA frequency by a factor of 10 was not sufficiently supported by the Section 7 analyses to be used in this value-impact analysis. Statistical arguments in NUREG-1050 (E.4) suggest that differences in magnitude by a factor of 3 from the historical data for LOCA are the threshold of discriminating between two parameters with large uncertainty. Such differences have not been observed as a result of IGSCC in BWRs. Engineering judgement can be used to conclude that the data base is sufficiently large to have included IGSCC and that only the availability of new methods in failure analysis allows the understanding of the specific mechanism involved in IGSCC. Thus, the sensitivity case of the factor of 3 increase was assumed for the high estimate in this analysis. The Gaertner report lists an average core melt increase for this case of 3.6 E-6 events per reactor-year excluding Big Rock Point. Big Rock Point was excluded because of differences in age, design, and siting when compared to the "typical" plant. This assumption is believed to have no significant effect on the plant average risk used in this analysis. Inclusion of Big Rock Point would distort results of this report because of the simplified approach to risk assessment that was used. The best estimate and low estimates used in this analysis assumed that the historical data are a good predictor of IGSCC contributions to LOCA and that the contribution is between 25% and 50% of the published values. This corresponds to reductions in core melt frequency of 9.0 E-7 per reactor year for the best estimate and 4.5 E-7 per reactor year for the low estimate.

To calculate the public risk impacts of the core melt frequency changes, they were multiplied by average dose factors generated using the CRAC code and assuming the quantities of radioactive isotopes and guidelines in WASH-1400, the meteorology at a typical midwestern site (Byron-Braidwood), a uniform population density of 340 people per square mile (an average of all U. S. nuclear power plant sites), and no evacuation of population. They are based on a 50-mile release radius model. These calculations result in 6.7 E+6 man-rem per core melt accident based on the Grand Gulf accident frequencies (E.1). An assessment of the validity of this assumption was made by determining the

LOCA contribution to each release category in the WASH-1400 Peach Bottom risk assessment. The contributions ranged from 3.8% to 4.7%. This suggests that use of an average dose factor does not induce large distortions in the risk results. Multiplying the dose factor result by the core melt frequencies predicts the following public risk values:

IGSCC Risk Contribution (man-rem per plant)

Low Estimate	76
Best Estimate	151
High Estimate	605

No differentiation for the various mitigation scenarios was made owing to the relatively low levels of risk value for mitigation of IGSCC. It was assumed that any of the various mitigation techniques would reduce LOCA frequencies due to IGSCC to negligible levels.

E.3.2 Occupational Exposure - Accidental

Reduced accidental occupational exposure could result from reducing the core melt frequency due to IGSCC. This value is believed to be small owing to the relatively low core melt reduction and was not estimated.

E.3.3 Public Property

Reduced public property loss could result from reducing the core melt frequency due to IGSCC. This value was not estimated owing to the relatively low core melt frequency reduction.

E.3.4 Onsite Property

Reduced onsite property loss could result from reducing the core melt frequency due to IGSCC. Value would result in the avoidance of core melt accidents and the reduction of the expected loss of onsite property. This value is believed low and was not estimated.

E.3.5 Occupational Exposure - Operational

Occupational exposure results from the implementation and operation of the three scenarios. The impacts are scenario dependent and are discussed below:

Scenario 1 - Pipe Replacement

The ORE for decontamination, replacement, and inspection of piping is made up of two contributors. These include the initial replacement and inspection activities and the inspection during resumed operations. The initial ORE is estimated directly from Section E.2.

Initial Replacement and Inspection ORE (man-rem)

Low Estimate	1500
Best Estimate	1750
High Estimate	2000

Addition of the HWC was assumed to add 20 man-rem during installation.

Inspection during the remaining life of the plant will be performed per the scope of ASME Section XI. This requires that 25% of the welds be inspected over a 10 year period. This scenario assumes that this inspection level will be met with or without HWC. Thus, the dose for operation of this resolution will be the unit value doses for inspection from Section E.2 multiplied by 75 welds. Results are as follows:

	Inspection ORE (man-rem)
Low Estimate	20
Best Estimate	60
High Estimate	225

Total ORE for this scenario with and without HWC is as follows:

	Total ORE (man-rem)	
	With HWC	Without HWC
Low Estimate	1540	1520
Best Estimate	1830	1810
High Estimate	2245	2225

Operation of the HWC has the potential to expose personnel in the area of the turbine during operation. Experience to date at Dresden 2 with HWC indicates that main steam gamma activity at full power increases by a factor of 5 because of increased 16N in the steam. However, the increase in steam activity disappears immediately (approximately 30 seconds) after hydrogen injection to feedwater is stopped. Therefore, shutting off the injection a short time before entering the proximity of the turbine or steam piping for short inspection/maintenance tasks will reduce the radiation field to negligible levels. For longer tasks, plant shutdown would be required regardless of HWC. It was assumed for this analysis that the HWC system is operated to minimize ORE. No additional ORE was added to account for this additional field in the plant.

Scenario 2 - IHSI

ORE in this scenario results from the implementation of the IHSI process, inspection of welds, and repair of welds over the remaining life of the plant. Implementation dose was estimated to range from 450 to 740 man-rem with a best estimate of 590 man-rem. Inspection dose was predicated on an 8% inspection rate for the first four outages after IHSI treatment if no HWC is used and inspection at the same level as pipe replacement without HWC if HWC is used. A total of 120 weld inspections would be required without HWC and a total of 100 weld inspections with HWC over the remaining 25 years of plant operation.

Repairs were assumed to be negligible if HWC is used. A total of five repairs were determined to be needed if HWC is not used. The inspection and repair frequencies were multiplied by the unit values in Section E.2 to generate estimates of ORE in the scenario. Results are as follows:

IHSI ORE (man-rem)

	IHSI	With HWC*		Without HWC		
	Implement	Inspection	Total	Inspection	Repair	Total
Low Estimate	450	30	500	36	12	498
Best Estimate	590	80	690	96	52	738
High Estimate	740	300	1060	360	98	1198

*Includes 20 man-rem for installation of HWC system.

Scenario 3 - Inspect and Repair

The inspect-and-repair scenario has inspection and repair dose when used with or without HWC. An additional 20 man-rem would be incurred during the installation of the HWC equipment.

To estimate the number of inspections and repairs required for this scenario over the remaining life of the plant, inspection results over all remaining outages were simulated. In all cases, there was an inititial period of increasing repair activity that declines as the population of weld repairs increases. Inspection activity shows a steady increase over the remaining life of the plant due to repeated inspection of the overlayed welds. For the purposes of ORE and cost calculations that are presented in the next section, average inspection and repair rates were established on an annual basis. This was calculated by summing the inspections and repairs required over the remaining life and dividing by the remaining plant life. Results with and without HWC are shown in the following table:

	Years	Insp. Rate, %	Average Annual Activity Inspections/yr	Repairs/yr
No HWC				
Low Estimate	1-9	40	49	3.6
	10-25	40	62	2.7
Best Estimate	1-9	50	60	4.4
	10-25	50	72	3.0
High Estimate	1-9	60	71	5.2
	10-25	60	81	3.3
With HWC				
Low Estimate	1-25	20	26	0.9
Best Estimate	1-25	30	38	1.3
High Estimate	1-25	40	49	1.7

These values were multiplied by their corresponding unit values from Section E.2 and the remaining plant life (25 years) to calculate the total ORE commitment for this scenario. Results are as follows:

ORE for Inspect and Repair Scenario (man-rem)

	With HWC	Without HWC
Low Estimate	249	611
Best Estimate	1094	2255
High Estimate	4504	7747

E.3.6 Industry Implementation Cost

Implementation costs for the three scenarios are developed in this section. They vary by scenario and are discussed below:

Scenario 1 - Pipe Replacement

Implementation costs for this scenario are due to the hardware removal and replacement and extended outages. The values for estimated outage time and other costs for pipe replacement were estimated in Section E.2. Replacement power was valued for this analysis at $300K/day. The total implementation cost for this scenario is as follows:

Pipe Replacement Implementation Cost ($ millions)

	Power Replacement	Other Costs	Total
Low Estimate	27	35	62
Best Estimate	41	44	85
High Estimate	68	57	125

Scenario 2 - IHSI

Implementation costs and outage times for IHSI were discussed in Section E.2. Conversion of the outage times at $300K/day was done as in Scenario 2. Results are as follows:

IHSI Implementation Costs ($ million)

	IHSI Treatment	Power	Other*	Total
No HWC				
Low Estimate	2.9	0	0	2.9
Best Estimate	7.0	23	0	30
High Estimate	10.9	32	0	43
With HWC				
Low Estimate	2.9	0	1	4
Best Estimate	7.0	23	2	32
High Estimate	10.9	32	3	46

*Implementation of HWC would add $1-$3 million with a best estimate of $2 million.

Scenario 3 - Inspect and Repair

This scenario without HWC has no implementation costs. HWC implementation costs are shown above.

E.3.7 Industry Operation and Maintenance Costs

This section discusses costs incurred following the implementation of the resolution to IGSCC. Each scenario is different and is discussed below. All scenarios contain discounted cash flow analyses at rates of 5% and 10% to estimate the present value of the activity.

Scenario 1 - Pipe Replacement

Operation and maintenance costs for this scenario result from inspection activity. The level of inspection was calculated on the basis of the requirements described for mitigation of IGSCC and ASME Section XI. Utilities were assumed to delay inspections as long as they could and still meet the code requirements. The inspection schedule used for this analysis was as follows:

	% Inspected		No. of Welds Inspected	
Years	HWC	No HWC	HWC	No HWC
1.5	0	4	0	5
3	8	4	10	5
6	8	8	10	10
9	9	9	10	10
12	8	8	10	10
16.5	8	8	10	10
19.5	9	9	10	10
22.5	8	8	10	10

Each of these inspections was costed using the unit values from Section E.2. The HWC system was assumed to have operating costs of $100K to $500K per year with a best estimate of $300K for 25 years. Present value results are as follows:

Present Value Operation Cost ($M)

	Low Estimate 5%	Low Estimate 10%	Best Estimate 5%	Best Estimate 10%	High Estimate 5%	High Estimate 10%
Inspection						
HWC	0.088	0.057	0.12	0.078	0.228	0.148
No HWC	0.088	0.057	0.12	0.078	0.228	0.148
HWC Operation	7.05	4.54	4.23	2.72	1.417	0.908
Total						
HWC	7.1	4.6	4.3	2.8	1.6	1.06
No HWC	0.088	0.057	0.12	0.078	0.228	0.148

Scenario 2 - IHSI

Operating costs for IHSI with HWC were assumed to be the sum of inspection costs for pipe replacement without HWC adjusted for 160 welds and the HWC operating costs. Costs for IHSI without water chemistry were developed assuming that repairs and inspections were conducted under the following schedule:

Years	No. of Welds Inspected	No. of Welds Repaired
1.5	13	1
3	13	-
4.5	13	1
6	13	-
12	13	1
16.5	13	-
19.5	13	1
22.5	13	1

The costs for inspection, repair, outage time, and HWC operating costs were estimated using the unit values from Section E.2. Present value operating costs for this scenario are as follows:

	Present Value Operating cost ($M) - IHSI	
	5%	10%
IHSI with HWC		
Low Estimate	7.3	4.7
Best Estimate	4.4	2.8
High Estimate	1.7	1.1
IHSI without HWC		
Low Estimate	0.39	.27
Best Estimate	6.8	4.8
High Estimate	13.3	9.3

Scenario 3 - Inspect and Repair

This scenario has operating costs due to inspection, repair, extended outage time and HWC operating costs. Estimates for outage time, inspection cost, repair cost, and HWC operating costs were developed in Section E.2. Outage time was costed at $300K per day. The estimates for required repair and inspection levels are time dependent. Averages on an annual basis were developed in Section E.3.5 to calculate the operational ORE. These values were multiplied by the Section E.2 unit cost values, added to the outage and HWC (if used) costs and discounted at 5% and 10% to generate the following present value estimates for the inspect-repair operational costs.

	Present Value Inspect and Repair Operational Costs ($M)	
	5%	10%
No HWC		
Low Estimate	85	56
Best Estimate	125	82
High Estimate	167	109
With HWC		
Low Estimate	36	24
Best Estimate	63	43
High Estimate	95	63

E.3.8 NRC Implementation Support Costs

Not estimated - believed to be relatively minor.

E.3.9 NRC Operational Support Costs

Not estimated - believed to be relatively minor.

E.4 REFERENCES

E.1 W. B. Andrews et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Priorization Information Development. NUREG/CR-2800 (PNL-4297), Pacific Northwest Laboratory, Richland, Washington.

E.2 Gaertner et al. 1983. Sensitivity of Core-Melt Frequency to LOCA Initiator Frequency for BWRs. Electric Power Research Institute, Palo Alto, California.

E.3 NRC. 1984. T. P. Speis to R. H. Vollmer. Memorandum dated January 6, 1984. "Briefing on BWR Pipe Crack Issues."

E.4 "Probabilistic Risk Assessment (PRA): Status Report and Guidance for Regulatory Application." Draft Report, NUREG-1050.

APPENDIX F

FRACTURE MECHANICS EVALUATION OF BWR RECIRCULATION PIPING

F.1 INTRODUCTION

Over the last ten years, intergranular stress-corrosion cracking (IGSCC) has been found in various stainless steel piping systems at operating boiling water reactor (BWR) facilities. During this time a number of fracture mechanics evaluations (F.1 through F.7) have been performed to evaluate the integrity of piping containing IGSCC and to determine the margins against unstable crack extension in the cracked piping.

Generally, the previous piping evaluations have been performed using either limit load analysis or tearing stability analysis based on an assumed fully plastic condition at the crack section. However, recent experimental data for stainless steel weld materials indicate a reduced resistance to crack extension compared to wrought stainless steel base metal. These data suggest toughness characteristics that may preclude cracked pipe sections reaching limit load prior to significant crack extension; consequently, analysis methods based on achieving limit load may not be strictly applicable for evaluation of flaws in stainless steel piping welds.

In view of the recent crack extension resistance data obtained for stainless steel weld materials, elastic plastic fracture mechanics analysis that do not necessarily assume limit load conditions at the cracked section were performed to assess the following: (1) the margin against unstable crack extension for finite length, deep part-through surface cracks, (2) the margin against unstable crack extension for throughwall cracks, and (3) the effect of variation in material toughness and tensile properties on the margin against unstable crack extension.

The analysis methods and the computational results are presented in the following section of this Appendix.

F.2 METHODOLOGY

F.2.1 General Considerations

The approach used to assess the integrity of cracked piping and the margins against unstable crack extension included performing the following analyses. First, part-through flaws, which are representative of the largest detected during an inservice inspection and allowed to remain in service, were evaluated to determine if they would remain stable for postulated faulted load conditions. Second, an evaluation was performed to determine the margin against unstable crack extension at faulted load conditions for a throughwall crack size corresponding to leak rates that can reliably be detected during normal operation by existing, in plant leakage detection equipment. Demonstrating adequate margins against unstable crack extension for these flaw conditions would indicate that leak-before-break conditions would be maintained and that there is little potential for pipe break.

Because there are questions about the accuracy with which part-through flaws can be detected or sized (see Chapter 4), additional throughwall flaw analyses were performed. These analyses included leak rate computations to determine the flaw sizes that produce leakage rates (at faulted loads) equal to the recirculation system makeup capacity, and stability analyses to define the flaw sizes resulting in unstable crack extension at faulted loads.

The fracture mechanics evaluation involves the following steps: (1) defining appropriately combined Service Level A and SSE loads, (2) choosing a suitably conservative flaw representation for the fracture mechanics analysis, (3) defining lower bound material properties to represent material crack initiation and growth resistance, (4) performing fracture mechanics computations to determine the potential for crack extension and the stability of subsequent crack growth, and (5) assessing the margins against unstable crack extension and net section plastic collapse in the cracked pipe section model. The following paragraphs in this section provide a detailed description of each of the variables used in the analysis and the analysis methods.

F.2.2 Loads

To perform the fracture mechanics and limit load analyses of the piping systems, it is necessary to identify the axial forces and bending moments acting on a section of pipe containing assumed circumferential cracks. The evaluations were made using combinations of Service Level A (SLA) and Safe Shutdown Earthquake (SSE) loads. The combined SLA and SSE loads were used because they realistically model the load condition of interest to demonstrate adequate margins against full break, i.e., an SSE occurs while the plant is at normal full power operation.

Service Level A loads include contributions from pressure, dead weight, and thermal expansion. However, thermal loads can either increase or decrease the resultant applied load; consequently, the thermal loads are included only if they result in a larger combined load. In general, this combined loading scheme is somewhat more conservative than Service Level D loads, which do not include thermal components. The system thermal expansion stresses are treated as primary stresses in the analyses. The contribution to crack extension potential from through-the-pipe-thickness stresses (resulting from thermal and residual stresses) are relatively small and, for the purpose of the stability analyses, were neglected in view of the conservative use of the pipe system thermal expansion stresses.

Two load classifications were used in the analysis. In one case SLA and SSE loads considered typical of most welds in the BWR recirculation system were used. A second group of analyses was performed using bounding loads that represent the highest loads at any of the welds in the recirculation system.

F.2.3 Flaw Characterization

F.2.3.1 Part-Through Flaws

Part-through flaws were evaluated to determine the margins against unstable crack extension for piping containing either undetected IGSCC or IGSCC that was found during inservice inspection and allowed to remain in service. The flaws used in the analyses are finite length, semi-elliptical circumferential cracks that extend radially into the pipe wall from the inside surface of the pipe.

The part-through circumferential flaw length was 50 percent of the pipe circumference. Flaw depths of 25, 50 and 75 percent of wall thickness were evaluated. The 50 percent flaw length and 50 percent depth are believed to bound the largest respective flaw dimensions that were returned to service following detection during inservice inspection. The 25 and 75 percent of wall thickness flaws were selected to obtain information about the effect of flaw depth on the potential for unstable crack extension.

F.2.3.2 Throughwall Flaws

In addition to ensuring adequate margins against unstable crack extension for part-through flaws, the evaluation of cracked piping generally includes demonstration that leak-before-break conditions will be maintained during postulated loading. This is important because, as discussed in the next section, cracks may grow throughwall by various time-dependent growth mechanisms, such as corrosion or fatigue. In this instance, a crack large enough to be detected by the leakage system should be able to withstand postulated loads without unstable crack extension. This assures that there is sufficient time to take action to detect and repair or replace piping with throughwall cracks.

The flaw selected to demonstrate leak-before-break is a flaw whose length corresponds to a leak rate that can reliably detected by leakage detection capabilities that meet the guidelines of Regulatory Guide 1.45 (F.8). A leak rate of 10 gpm is used to define the throughwall flaw length and to ensure detection with some margin to cover analytical and measurement uncertainties.

A correlation between crack size and flow rate through the crack was calculated using a computer code previously developed for the Electric Power Research Institute (F.9). Computational procedures in this code are based on two-phase flow through a cracklike opening and consider crack face surface roughness, crack length and wall thickness, pressure differential across the crack, and crack opening area due to the loading. The crack opening area and associated leak rate are determined using normal operating (SLA) load conditions.

In addition to the throughwall flaw condition just described, leak rate and fracture mechanics analyses were performed to determine two additional throughwall crack sizes. First, leak rate analyses were performed to find the throughwall flaw size associated with a leak rate equal to the makeup capability (about 100 gpm) for the recirculation system at SLA plus SSE loads. Second, fracture mechanics analyses were performed to determine the throughwall crack size that would result in unstable crack extension at SLA + SSE loads. These two flaw sizes serve to demonstrate the added margin for throughwall crack size relative to the size that can be detected reliably at SLA conditions by existing in-plant leakage detection systems.

F.2.4 Throughwall Crack Development

Because part-through flaws will not be found by leakage detection, definition of the flaw used in the evaluation also considered the effect of part-through flaws on leak-before-break conditions to demonstrate that the throughwall flaw is an appropriate bounding flaw for analysis purposes. The purpose of this section is to demonstrate that part-through cracks are likely to penetrate the pipe wall thickness and leak rather than progress around the pipe and cause a significant break.

This crack growth sequence is verified for the two conditions that are of major interest; namely, normal operation, and large bending loads in excess of those postulated for faulted (SSE) loading.

For normal operating conditions, there is a large amount of service experience that demonstrates that stress-corrosion cracks progress radially through the pipe wall and result in leak-before-break conditions. As indicated in F.2, F.10, and F.11, several hundred significant cracking incidents have been discovered in the United States in BWR primary and secondary piping systems, and PWR secondary piping systems. These statistics represent experience with a wide range of crack sizes and piping systems. These cracks result from various initiation and propagation mechanisms, and exposure to different combinations of stress states, i.e., bending and tension. For all the different conditions that actually occur in service, operating service experience indicates that the dominant crack growth trend for intermediate- and larger-diameter piping is for the crack to grow radially through the wall and leak.

Because large earthquake loadings occur very rarely, service experience is not available to define the character of crack growth for these loads. However, results from previously performed analyses do provide an indication of part-through crack characteristics for large bending loads. The studies described in F.5 defined the ratio of the crack driving force (in terms of the J integral) in the circumferential-to-radial direction for a part-through crack in a pipe. The analytical results indicate that the value of the crack driving potential J in the radial direction, J_a, is always greater than the value of the J in the circumferential direction, J_ϕ, for all combinations of percentage of crack through the wall, a/t, and circumferential distance around the pipe. This variation is shown in Figure F.1, which is taken from F.5. The variation of J_ϕ with J_a shown in Figure F.1 demonstrates that part-through cracks will grow through the pipe wall and leak when the pipe is subjected to large postulated bending (SSE) loads.

From service experience and analysis, it can be concluded that it is likely cracks will grow radially through the pipe wall before they progress around the pipe and cause a significant break. This phenomenon has been demonstrated for operating conditions ranging from normal operation to large accident loads, and for a wide range of tensile and bending loading combinations. This ensures that analyses based on large circumferentially oriented, throughwall flaws are an appropriately conservative basis for evaluating the margin against full break and ensuring leak before break conditions are maintained.

F.2.5 Fracture Mechanics Evaluation

F.2.5.1 General Considerations

Elastic-plastic fracture mechanics analyses were performed for SLA and loads in excess of SSE loads to evaluate the potential for unstable crack extension of through and part-through flaws. The flaw geometries used in the analyses are illustrated in Figure F.2. The methods used appropriately account for material yielding and strain hardening that may occur in the cracked section on application of large loads, and/or the presence of ductile crack growth, which is the expected crack extension mechanism for ductile nuclear reactor pipe materials.

The analyses are based on a parameter called the J integral (F.12), which is a measure of the intensity of the stress-strain field around the crack tip (F.13, F.14). The two important aspects should be considered in general when evaluating crack growth; namely, initiation or first extension of the existing flaw and stability of a growing flaw subsequent to initiation. The material value of J associated with initiation of additional crack extension is denoted as J_{IC}. If the applied value of J is less than J_{IC}, no crack growth will occur and stability of the existing crack is ensured automatically. When extension of the existing crack is predicted, the crack extension must be evaluated to determine if it occurs in a stable manner, or if the crack will grow unstably and result in a predicted full break.

Crack instability is evaluated using the tearing stability concept and the associated tearing modulus instability criterion (F.15). Tearing modulus is defined as

$$T = \frac{dJ}{da} \frac{E}{\sigma_f^2} \quad , \tag{1}$$

where

dJ/da	indicates the increment of J needed to produce a specified increment of crack extension at any specified load and crack state,
E	is the material elastic modulus, and
σ_f	is the material flow stress defined as one half the sum of material yield and ultimate strengths.

To determine the margin against fracture, the values of J and T are determined for the structure using the applied loads and crack geometry. The values are denoted as $J_{applied}$ or J_{app}, and $T_{applied}$ or T_{app}.

The material resistance to unstable crack extension is determined experimentally from test data that relate J and crack extension. Using these data the tearing modulus can be determined from the slope of the crack growth curve using Eq. (1). The tearing modulus obtained from the materials data is denoted as T_{mat}. At any specified J level, where J_{app} is greater than J_{IC}, stable crack extension will occur when

$$T_{mat} > T_{app} \quad . \tag{2}$$

A convenient means now commonly used to illustrate the margin against instability involves plotting J as a function of T for the applied loads and material resistance as shown schematically in Figure F.3. In this figure material crack growth resistance is plotted along with the crack growth potential associated with the load and crack in the component. The intersection of the curves defines the instability point.

F.2.5.2 Computation of J_{app} and T_{app}

The computation of J follows the method described in F.16 and F.17 for through flaws. The same general formulation also was used in this study to obtain J for part-through finite-length semi-elliptical surface flaws. In this generalized computation scheme the J integral is separated into elastic and plastic components, or

$$J = J_e + J_p \, , \tag{3}$$

where

J_e is the plasticity adjusted elastic contribution to J

and

J_p is the plastic contribution to J.

The elastic portion of J is directly related to the elastic stress intensity factor, K_I, by the relationship

$$J_e = K_I^2/E. \tag{4}$$

Elastic K solutions are available from F.18, F.19, and F.20.

The plastic component of J for (bend loading) is expressed in the form (F.17).

$$J_p = \alpha\, \sigma_o \varepsilon_o c h_1 (M/M_o)^{n+1} \quad , \tag{5}$$

where

σ_o and ε_o are the reference yield stress and strain and α and n are material constants determined from the material stress strain curve fit to a Ramberg-Osgood curve.*

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* $\varepsilon/\varepsilon_o = \sigma/\sigma_o + \alpha(\sigma/\sigma_o)^n$.

2c is the remaining circumferential ligament of the cracked portion of the pipe;

h_1 is a function which accounts for relative crack and component size, and material work hardening;

M_o is the moment required to develop an average stress of magnitude σ_o in the cracked section;

M is the applied moment.

The total J is obtained by adding the plasticity adjusted elastic solution and the plastic solution given by equation 5, or

$$J = J_e(a_{eff}) + J_P. \tag{6}$$

The applied tearing modulus can be expressed in general terms (for bending) (F.21) as

$$T_{app} = \frac{E}{\sigma_f^2}\left[\left(\frac{\partial J}{\partial a}\right)_M - \left(\frac{\partial J}{\partial M}\right)_a \left(\frac{\partial \phi_c}{\partial a}\right)_M \left\{C + \left(\frac{\partial \phi}{\partial M}\right)_a\right\}^{-1}\right], \tag{7}$$

where

a = crack size,

M = Moment,

ϕ = total angle of rotation,

ϕ_c = angle of rotation of cracked section, and

C = system compliance

The system compliance ranges between two extremes; $C = 0$ for fixed displacement boundary conditions and $C = \infty$ for dead weight type loading. A conservative estimate of T_{app} can be obtained by placing $C = \infty$ into Eq. (8) so that it becomes

$$T_{app} = \frac{dJ_{app}}{da} \frac{E}{\sigma_f^2} \quad . \tag{8}$$

F.2.5.3 Materials Data

Analyses were performed for two weld materials including submerged arc welds and tungsten inert gas welds. These two welds are believed to represent the majority of the welds in BWR recirculation piping.

The crack extension data for the materials are expressed in the form of applied J versus measured crack extension. To ensure that the material crack growth characteristics are used in a conservative manner, an attempt was made to represent the data as reasonable lower bounds.

All data were obtained from 1-in. - thick compact tension-type (ITCT) test specimens that had face grooves. The use of face grooves on the CT specimen is the configuration that appears best to simulate the cracked tip condition associated with cracked piping. The material test temperatures were about 550°F.

F.2.6 Net Section Plastic Collapse

The presence of the postulated circumferential throughwall flaw will reduce the ultimate load-carrying capacity of the pipe section.

The limit moment for the pipe section containing a throughwall crack was determined from F.1 and is expressed as

$$M_f = 4\sigma_f R^2 t \,(\cos\gamma - \tfrac{1}{2}\sin\theta) \quad , \tag{9}$$

where

R is the mean pipe radius,

t is the pipe wall thickness,

θ = half-crack angle, and

$$\gamma = \tfrac{1}{2}\theta + \frac{\pi}{2}\,\frac{\text{Axial load}}{2\pi R\sigma_f t} \ .$$

F.3 NUMERICAL RESULTS

Numerical computations were performed to assess several aspects related to the integrity of flawed piping. These aspects include: (1) the loads at instability relative to the limit load, (2) the margin against first extension of part-through cracks, (3) the margin against unstable crack extension for throughwall cracks, and (4) the effect of material tensile and toughness properties on the margin against first crack extension and unstable crack extension. The following paragraphs of this section present the numerical input used to complete this analyses and the computational results.

F.3.1 Pipe Dimensions

Selected pipes in the BWR "ring header" design recirculation piping system were evaluated. These pipes include the 28-in. pump suction and discharge lines, the 22-in. header, and the 12-in. risers. The pipe dimensions used in the analysis are presented in Table F.1.

Table F.1

Recirculation System Pipe Dimensions

Nominal Diameter in.	Mean Radius in.	Wall Thickness in.
12	6.092	0.566
22	10.475	1.05
28	13.45	1.1

F.3.2 Loads

Two loading conditions were used in the analyses. These included a bounding loading condition representative of the largest stress for each pipe size and a typical loading condition representative of the stresses at most weld locations in recirculation piping. The SLA and SSE stresses for each loading condition and pipe size are presented in Table F.2.

Table F.2

Recirculation System Typical and Bounding Load Definition

Nominal Pipe Size in.	Service Level A Stresses Axial Typical (ksi)	Service Level A Stresses Axial Bounding (ksi)	Service Level A Stresses Bending Typical (ksi)	Service Level A Stresses Bending Bounding (ksi)	SSE Stresses* Bending Typical (ksi)	SSE Stresses* Bending Bounding (ksi)
12	6	6	6	10	4	10
22	6	6	3	6	3	10
28	6	6	3	6	3	10

*No additional axial stress is attributed to seismic loading.

F.3.3 Throughwall Crack Size Versus Leak Rate Correlation

The flaw sizes associated with a 10-gpm and 100-gpm leak rate were obtained for each pipe size at both the typical and bounding load conditions. The 10-gpm crack size represents a leakage size crack that could reliably be detected at normal operation and is used to demonstrate leak-before-break margins relative to faulted loading. The 100-gpm crack size represents the crack size that would approximately equal the makeup capacity for the recirculation system at faulted loading. The results are presented in Table F.3.

Table F.3

Detectable Leakage and Makeup Capacity Size Throughwall Flaws for Typical and Bounding Loading

Nominal Pipe Size in.	10-gpm Crack Size Relative to Circumference Service Level A Load		100-gpm Crack Size Relative to Circumference SLA + SSE Load	
	Typical θ/π	Bounding	Typical θ/π	Bounding
12	0.21	0.19	0.35	0.24
22	0.16	0.14	0.30	0.21
28	0.14	0.12	0.26	0.18

F.3.4 Fracture Mechanics Evaluation

F.3.4.1 Materials Data

F.3.4.1.1 Tensile Properties

The tensile properties for each of the materials are presented in Table F.4.

Table F.4

Tensile Properties for Stainless Steel Weld Materials, Temperature = 550°F

Property	Material: Submerged Arc Weld	Material: Tungsten Inert Gas Weld
Yield Strength, ksi	53.9	49
Ultimate Strength, ksi	63.4	65.6
Flow Stress, ksi	58.7	57.3
Reference Stress, ksi	53.9	35
α	2.83	2.52
N	11.8	5.0

F.3.4.1.2 <u>Toughness Properties</u>

The toughness properties for the two weld metals listed in Table F.4 were obtained from experiments that generated J versus crack extension (J-R) curves at a temperature of 550°F. The J-R curves provide two measures of fracture resistance; namely, resistance to first extension of the existing crack (J_{IC}) and the resistance to a growing crack.

The values of J_{IC} for each material are presented in Table F.5. Generally, an assessment of stability conditions will be based on the resistance to crack extension of a growing crack. As discussed in Section F.2.5.1, it is convenient to construct a J/T plot from the J-R curve data for performing the stability analysis. The J/T curve for each material is shown in Figure F.4. The J-R curves used to construct the J/T plots are presented in Figures F.5 and F.6. The data presented in Tables F.4 and F.5 and Figures F.4, F.5 and F.6 were obtained from ongoing NRC and EPRI programs (F.22, F.23).

Table F.5

J_{IC} Values at 550°F for Submerged Arc and Shielded Manual Metal Arc Weld Metals

Material	J_{IC} in.-lb/in.^{2}
Submerged Arc Weld (SA)	420
Tungsten Inert Gas Weld (TIG)	960

In most instances information presented in Tables F.4 and F.5 and Figure F.3 were obtained from tensile and toughness tests from one or two specimens. The tensile data presented in Table F.4 represent the average measured values and are appropriate for use in the equations presented in Section F.2.5.2. The tensile data in Table F.4 for the submerged arc weld were not measured directly but were estimated by the experimental investigators based on properties from other welds (F.24).

The J/T plots generally are constructed using the lower bound of the available J-R data and are appropriate for obtaining a realistically conservative estimate of the instability conditions.

F.3.4.2 <u>Numerical Results</u>

F.3.4.2.1 <u>Part-Through Surface Cracks</u>

Values of J were computed for several loads and flaw sizes. The load levels included the bounding SLA, SLA + 1SSE, and SLA + 2SSE loads. The flaw circumferential length was assumed to be 180° around the pipe, while the radial part-through flaw depth was taken as 25, 50, and 75 percent of the pipe wall thickness. Computations were performed for each of the three pipe sizes and the submerged arc and tungsten inert gas weld materials.

The results obtained for the submerged arc weld indicate that J is less than J_{IC} (420 in.-lb/in.^{2}) for the loads, flaw sizes, and pipe sizes evalated, with the exception of a 75 percent part-through flaw in a 28-in. pipe subjected to SLA + 2SSE loads. In this case the computed J is 700 in.-lb/in.^{2}. Because J is greater than J_{IC}, the value of applied tearing modulus, T, was determined to assess if the predicted crack growth is stable. A comparison of the computed applied values of J = 700 and T = 20 with the J/T plot in Figure F.4 for the submerged arc material indicates stable crack growth.

Similar computations were made for the tungsten inert gas weld material. The results of these analyses indicate applied J values less than J_{IC} (960 in.-lb/in.^{2}) for all three pipe sizes with SLA + 2SSE loads and flaw depths of 50% of the wall thickness. At flaw depths of 75% of the wall thickness applied, J was greater than J_{IC} for all three pipe sizes at SLA + 2SSE loads. The computed applied J values are 1450, 1370, and 1990 in.-lb/in.^{2} for the 12-, 22-, and 28-in.-diameter pipes, respectively. The respective applied T values for each pipe size were determined to be 260, 133, and 165. Comparison of these values with the plot in Figure F.4 for the tungsten inert gas weld indicates stable crack extension for the 180°-long, 75 percent part-through flaws at SLA + 2SSE loads.

F.3.4.2.2 Throughwall Cracks

This section presents the results of the computation performed to determine the instability moments for the 10-gpm leakage crack sizes associated with the bounding SLA loads (see Table F.3), and the instability crack sizes at SLA + 1SSE loads.

The instability moments for the leakage size cracks in the 12-, 22-, and 28-in. pipes are presented in Table F.6. Also presented in the table is the ratio of the instability moment to the limit moment for the two weld materials.

Table F.6

Instability Moment for Bounding Load SLA

Leakage Size (10 gpm) Cracks

Pipe Diameter in.	Leakage Crack Size Relative to Circumference	Instability Moment 10^6 in.-lb		M_I/M_L[1]	
	θ/π	SA[2]	TIG[3]	SA	TIG
12	0.19	2.5	2.2	0.81	0.74
22	0.14	14.9	12.9	0.79	0.70
28	0.12	26.8	23.1	0.76	0.67

[1]Ratio instability moment to limit moment.

[2]SA = submerged arc weld.

[3]TIG = tungsten inert gas weld.

The results in Table F.6 indicate that the instability moment ranges from about 70 to 80 percent of the limit moment for the materials evaluated in this study. This result suggests that the use of limit load is not strictly appropriate for flaw evaluation. However, the instability moments are significantly larger than the SLA + 1SSE moments and significant margins against unstable crack extension exist as discussed in Section 3.5.

The throughwall crack lengths that would produce unstable crack extension at typical and bounding SLA + 1SSE loads were determined for the two weld materials. The results are presented in Table F.7.

Table F.7

Crack Size for Instability at SLA + 1SSE Loads

Pipe Diameter in.	Instability Crack Size Relative to Circumference at Typical SLA + 1SSE Load		Instability Crack Size Relative to Circumference at Bounding SLA + 1SSE Load	
	SA[1]	TIG[2]	SA	TIG
	θ/π		θ/π	
12	0.42	0.44	0.31	0.32
22	0.46	0.47	0.32	0.35
28	0.41	0.43	0.28	0.31

[1]SAA = submerged arc weld.

[2]TIG = tungsten inert gas weld.

The results in Table F.7 indicate that relatively large (30% around the circumference) throughwall flaws can be tolerated at the bounding SLA + 1SSE loads and somewhat larger sizes (40% around the circumference) for the typical SLA + 1SSE loads. The results are about the same for the two welds.

The results in Table F.7 also serve to point up the importance of tensile properties in the analyses. For example, the instability crack size for the two welds are about the same even though there is a significant difference in the J-R curves for the materials (see Figure F.4). The difference in J-R curves is compensated for in this case by the higher strength, flatter (lower work hardening) stress strain curve for the submerged arc weld compared to the tungsten inert gas arc weld.

F.3.5 Margins

F.3.5.1 Part-Through Wall Flaws

The margins against first crack extension or unstable crack extension can be estimated from the results in Section 3.4.2.1. Although the loads corresponding to unstable crack extension were not obtained, the margin against unstable crack extension can be estimated using the SLA + 2SSE load as a reference. At this load the flaws either did not extend or exhibit stable growth for each of the pipe sizes and materials evaluated. Consequently, realistically conservative estimates of margin at SLA and SLA + 1SSE load can be obtained by the ratio of $P_m + P_b$ at each of these conditions (see Table F.2) to the value of $P_m + P_b$ at SLA + 2SSE. The results are presented in Table F.8 for the bounding load case.

Table F.8

Estimated Margin Against Unstable Crack Extension for Part-Through Flaws Up to 75% of Wall for Submerged Arc and Tungsten Inert Gas Arc Welds

Pipe Diameter in.	$P_b + P_m$ at SLA + 2SSE ksi	Margin Relative to SLA + 2SSE Bounding Loads	
		SLA	SLA + 1SSE
12	36	2.3	1.4
22 & 28	32	2.7	1.5

The margins shown in Table 8 generally can be considered relatively close to the margins corresponding to instability margins for the tungsten inert gas weld as seen from the J/T values in Section 3.4.7.1. The margins would be somewhat higher for the submerged arc weld where the applied J and T values are typically lower for the 75 percent part-through flaw.

The results in this section also provide limited data that can be used to evaluate the margins in section IWR 3640 of Section XI of the ASME Code. Even though limit load is not achieved for the weld materials considered, margins equivalent to those in the code are achieved (for the loads used in this evaluation) because the part-through flaw depths allowed by IWB 3640 for a 180° flaw are less than the 75 percent of wall depth that was shown to remain stable by this analysis. To ensure this conclusion can be universally true, additional flaw depths and lengths and loading conditions representative of the scope of IWB 3640 would have to be evaluated.

F.3.5.2 Throughwall Flaws

The margins against instability for pipes with leakage size (10-gpm leak rate) throughwall flaws were determined using the instability results from Table F.6 and stress conditions in Table F.2. The results are presented in Table F.9 for the two weld materials and both the bounding and typical SLA + 1SSE loads.

Table F.9

Margin Against Instability for Total Load

($P_m + P_b$) at SLA + 1SSE Loads for

10 gpm Leakage Size Throughwall Flaws

Pipe Diameter in.	$P_m + P_b$ at Instability, ksi		Margin Against Instability at SLA + SSE			
			Typical Loads		Bounding Loads	
	SA	TIG	SA	TIG	SA	TIG
12	42.9	39.3	2.7	2.5	1.7	1.5
22	47.2	41.6	3.9	3.5	2.1	1.9
28	48.9	43.	4.1	3.6	2.2	2.0

The results in Table F.9 show significant margins against unstable crack extension for the bounding loads and large margins for the typical loads. These results indicate that most weld joints in the recirculation system have large margins for leak-before-break and that the highest stress weld joints have margins typical of values normally associated with ASME Code margins for faulted conditions.

In addition to determining margin on loads, margin on flaw size also can be defined by comparing the leakage size flaws in Table F.3 with the instability flaw sizes in Table F.7. From these data, the margin on the 10-gpm leakage size flaw generally exceeds a factor of 2.0 relative to the instability size, except for the 12-in.- diameter pipe where the margin is about 1.6. The data in the tables also indicate that the instability flaw size is always larger than the flaw size corresponding to the system makeup capacity of 100-gpm at SLA +1SSE.

F.4 CONCLUSIONS

Part-through cracks as large as 50 percent around the circumference and 75 percent throughwall will remain stable at loads in excess of SLA + 2SSE. This implies margins on total load of 1.4 and 2.3 relative to SLA + 1SSE and SLA, respectively, for the highest stressed weld joints in the recirculation system.

The margin against unstable crack extension at SLA + 1SSE for leakage size (10-gpm) cracks ranges from factors of 1.5 to 2.2 on load for the highest stressed weld joints in the recirculation system. The instability crack size generally exceeds the 10-gpm crack size by a factor of 2.0, except for 12-in. piping where the factor is about 1.6. The crack size at instability is always larger than the crack size corresponding to the makeup capacity of the recirculation system.

The margins on crack size and load indicate that leak-before-break conditions will be maintained for BWR recirculation piping.

On the basis of the limited results obtained from these analyses, it appears that the safety factors incorporated in paragraph IWB 3640 of Section XI of the ASME Code are generally maintained for BWR recirculation piping, even though crack instability is predicted prior to reaching load limit for the pipe sizes and welds evaluated.

The tensile properties play a large role in evaluating crack stability using elastic-plastic techniques. The results obtained in this analysis show that the relative difference in tensile properties between the submerged arc and tungsten inert gas welds almost completely compensates for the significant difference in the material J/T relationships.

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F.21 J. W. Hutchinson and P. C. Paris. 1979. Stability Analysis of J-Controlled Crack Growth. Am. Soc. Test. Mater. Spec. Tech. Publ. 668 37-64.

F.22 EPRI Program RP 1238-2, Westinghouse.

F.23 NRC Program FIN B-6290, David Taylor Naval Ship R&D Center.

F.24 Private Communication with D. McCabe, Westinghouse. March 13, 1984.

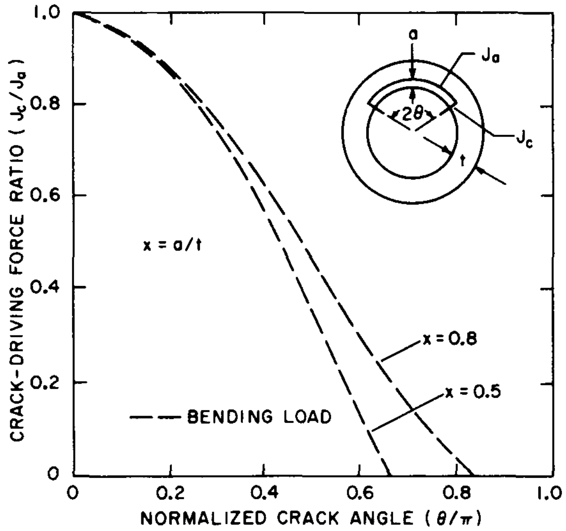


Figure F.1 Stability of a part-through circumferential crack under fully plastic bending loads.

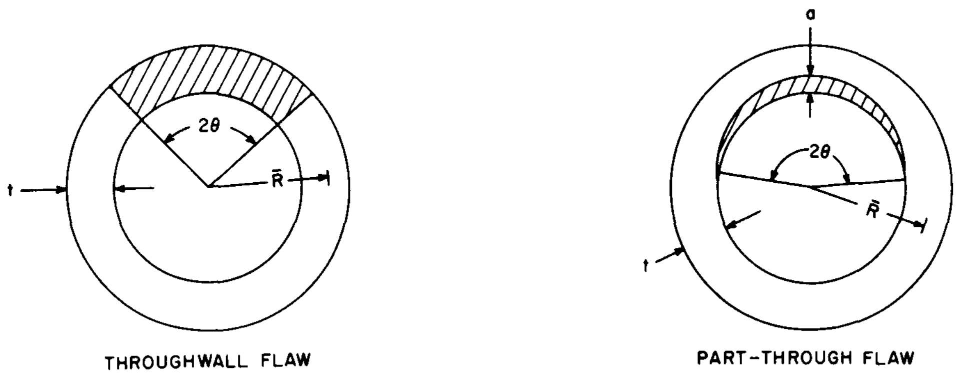


Figure F.2 Through and part-through geometries used in fracture mechanics analyses.

Figure F.3 Illustration of stability assessment diagram.

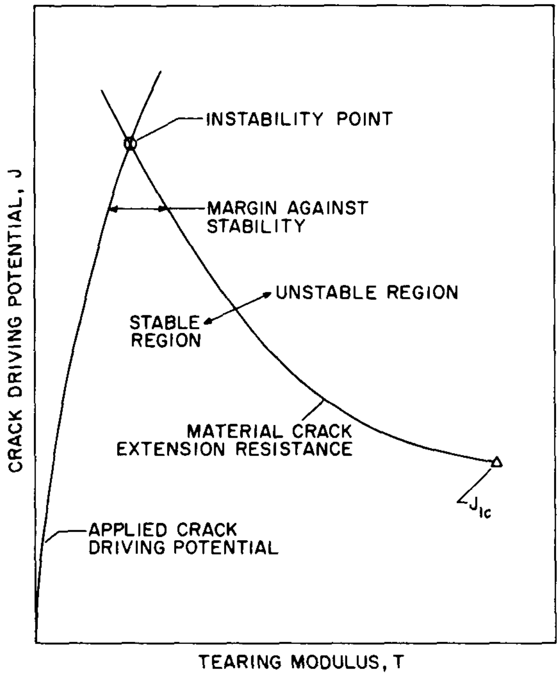
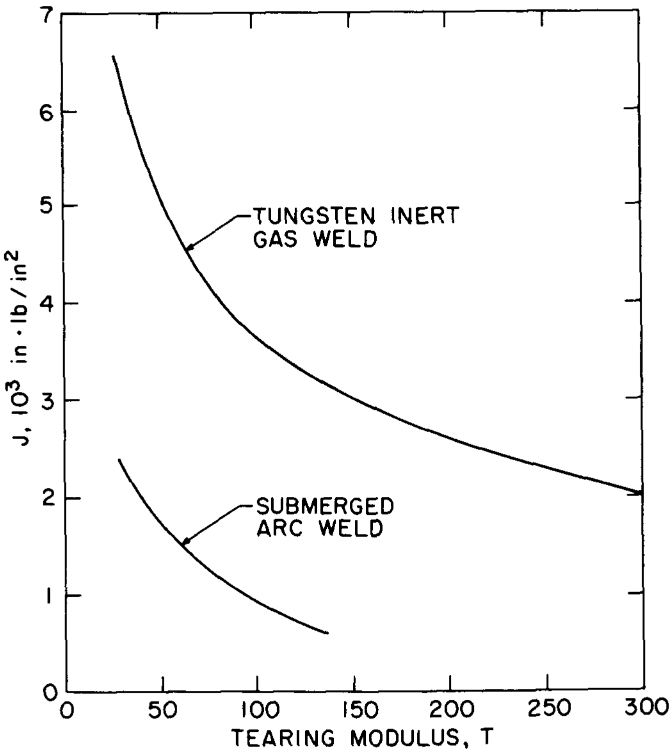


Figure F.4 J/T Plots for Submerged arc and Tungsten inert gas welds, 550°F.



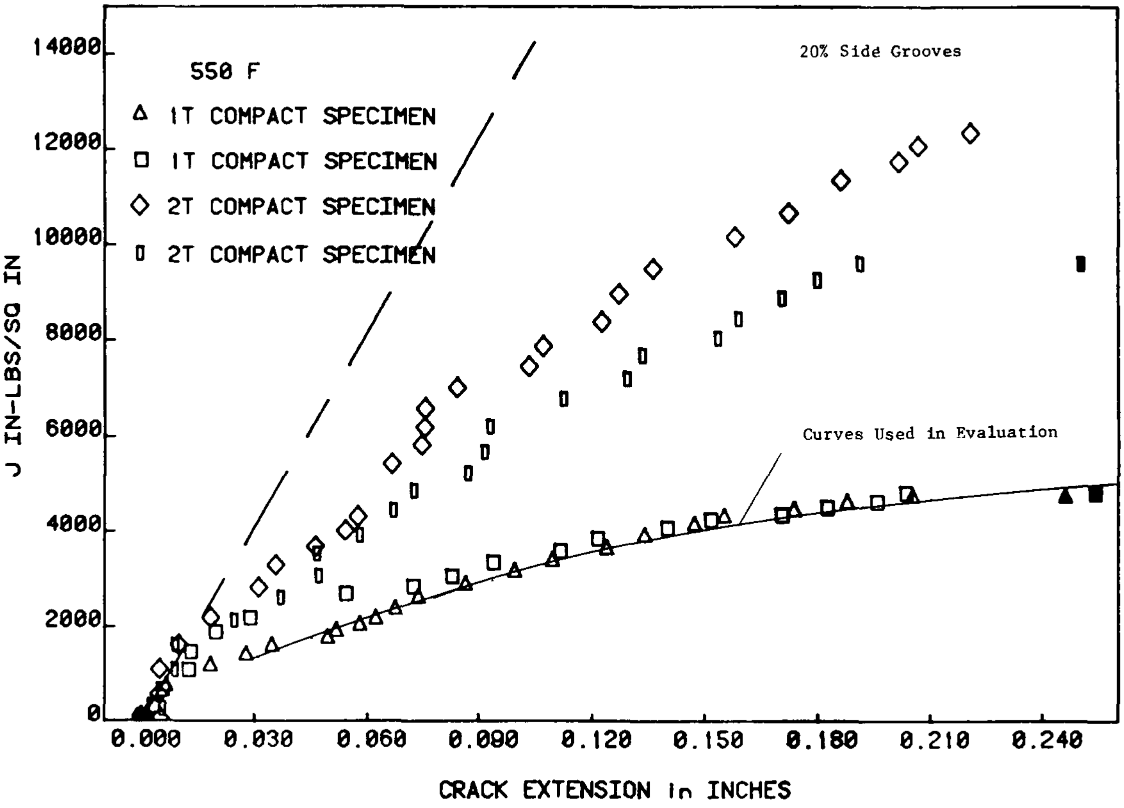


Figure F.5 J-R Curve for TIG Weld, 550°F (Ref. F.23)

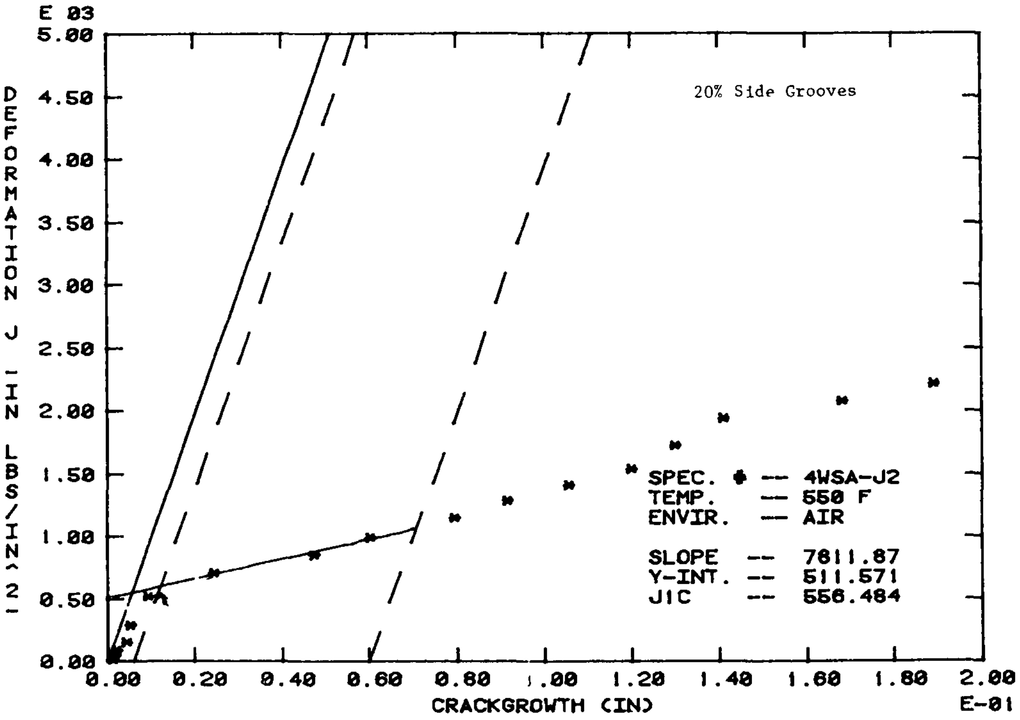


Figure F.6 J-R Curve for Submerged Arc Weld, 550°F (Ref. F.22)

APPENDIX G

LONG-TERM ISSUES

ULTRASONIC INSPECTABILITY

Substantial improvement in the detection of IGSCC during ISI of BWR piping using UT has been made. This improvement has been accomplished through a combination of operator training, elimination of ineffective procedures, and the implementation of improved procedures (techniques). The current crack detection performance level is considered acceptable only on an interim basis, and additional efforts are necessary to achieve further improvements on a long-term basis. Currently, programs are under way through NRC, ASME, and the industry to bring about these improvements.

Generally, the current capability for accurately sizing flaws, particularly flaw depths, is considered unsatisfactory. Length measurements are marginally acceptable. Therefore, better procedures are needed. Flaw sizing needs to be examined in the context of the failure probabilities resulting from a given flaw. For example, a short flaw even when 100% throughwall will not cause pipe failure. Longer flaws require greater care in measurement of both length and depth. Currently, programs are under way through NRC, ASME, and the industry to accomplish these improvements.

In the long term, both UT systems and piping systems should be designed to permit automatic examination of welds. This may require redesign or modification of weldments. Such a measure is essential to ensure as low as reasonably achievable (ALARA) personnel radiation exposure. Today beneficial improvements can be achieved by using current technology which automatically records UT signals and scanning (positional) data, although at this time this technology is not generally being used in the field.

In new plants and for pipe replacement, attention should be given to inspectability and accountability for inspection throughout the design stage. An objective of 100% inspectability for all primary system welds should be established. Inspectability is used to denote access to both sides of the weld, a minimization of signal-to-noise ratio, and a weldment optimized to permit UT examination. This consideration should apply equally to mono-, bi-, and trimetallic welds.

LEAK DETECTION

There should be a careful review of existing leak detection systems regarding their reliability and sensitivity now and after significant operating time. Also, attention should be given to the development of more sensitive leak detection and continuous crack growth monitoring systems aimed at detecting leakage with higher confidence and at lower leak rates.

REPLACEMENT MATERIALS

The suggested replacement material for 304 SS, namely 316NG, is a much superior material for BWR piping systems. The material is highly resistant to initiation and growth of IGSCC in BWR environments. This applies to properly fabricated welds as well as the base metal. Although extensive data exist, they are not sufficient to say the 316NG material will be completely resistant to cracking; however, the probability of cracking appears acceptably low. It is recognized that none of the suggested replacement materials are resistant to cracking in severe environments such as those resulting from chloride intrusions.

WATER CHEMISTRY

EPRI and NRC are sponsoring laboratory research and demonstration projects at operating plants to evaluate the effectiveness of changes in BWR water chemistry in the control of both the initiation and propagation of stress corrosion cracks in primary piping. These changes include tightening the controls and limits on intrusion of impurities from the demineralizers and the injection of hydrogen into the feedwater system to reduce the oxygen levels. Possible adverse effects of hydrogen injection on other parts of the system need to be evaluated from the demonstration projects before hydrogen injection can be recommended as a mitigating action. These include effects on radiation levels in the steam system, on the corrosion product transport mechanism, and on hydriding of fuel cladding.

FRACTURE ANALYSIS

Although operating experience suggests a low probabilty of catastrophic failure in austenitic stainless steel piping systems, certain questions remain to be answered before the magnitude of safety margins associated with leak-before-break can be assessed reliably for the desired range of BWR operating conditions. These questions generally concern material properties, flaw size, and analysis methodology.

Work to address each of these items, both in the short and long term, either has been initiated or will be initiated soon. Completion of this work will provide a comprehensive basis for the justification and use of deterministic evaluation techniques to define safety margins.

A combination of probabilistic studies, experimental testing and sophisticated analytical studies should be conducted to establish the relative probability of leak before break compared to catastrophic failure. These programs should address existing materials such as 304 SS and replacement materials such as 316NG. The NRC Degraded Piping Program and the probability fracture mechanics program at Lawrence Livermore National Laboratory are evaluating this issue. As the long-term goal, major sources of failure and system designs should be identified to eliminate such sources of failure. This will require the joint efforts of NRC, Code Groups, and industry.

MITIGATING MEASURES

Mitigating measures such as induction heating stress improvement (IHSI) and heat sink welding (HSW) are aimed at reducing or reversing inner surfce residual stresses, thus preventing or greatly extending the time for IGSCC initiation. Available information is considered adequate to make regulatory decisions for their use on new pipe; however, further work is required to establish limits to their use on pipe containing cracks. This work should consider a range of flaw depths to establish if long-term limitations exist with IHSI on pipes in operating plants. There appear to be no substantial negative effects related to use of IHSI or HSW.

Many plants have used weld overlay reinforcement on cracked pipes. Although this is currently considered to be an acceptble short-term (one fuel cycle) remedy, some utilities may find it expedient to continue to operate with overlayed welds for longer time periods. The main questions involved in the acceptability of longer-term operation with overlayed welds is the possibility of further crack extension and inspectability. These subjects will be addressed by the NRC and new EPRI programs.

IGSCC SUSCEPTIBILITY

IGSCC has been occurring in BWR piping systems for more than 20 years. Corrective actions have been taken over the past two decades to replace cracked weldments with low-carbon stainless steels, and these replacements have resisted cracking. However, even after replacement of some of the major piping systems, there will still remain a great deal of high-carbon sensitized stainless steel piping; attention should be given in the long term to replacing all weldments potentially susceptible to IGSCC.

Recirculation and other BWR piping systems in older plants were designed to less rigorous criteria with regard to loads and response to emergency or faulted conditions. A long-term goal tied to the replacement of systems susceptible to IGSCC should be an upgrading in overall design criteria as well as modification of system geometries to optimize accessibility to weldments and to minimize the number of weldments. These efforts should be directed to meeting new versions of regulatory positions where considered appropriate.

RISK-ASSESSMENT - VALUE-IMPACT ANALYSIS

BWR risk assessment studies, including WASH-1400, have concluded that large-break LOCAs are only minor contributors to the public risk. Continuing review of these studies and the data base are expected to show no change in this conclusion or the data on which it is based. Long-term BWR risk assessment is not expected to produce any different conclusions unless the data base changes radically with further maturation of the nuclear power plant population.

If no radical changes in risk assessment results occur, it is anticipated that value-impact analysis will involve evaluation of various IGSCC-mitigating schemes or corrective actions in terms of occupational radiation exposure (ORE) and utility costs.

SUMMARY

To ensure acceptable margins against IGSCC in BWR piping, utilization of all three methods beneficial in preventing or delaying IGSCC should be considered. These methods are replacing material with more resistant material, obtaining a beneficial stress distribution through IHSI or HSW, and minimizing corrosion initiators in the coolant.

APPENDIX H

April 6, 1984

Mr. Joseph D. Lafleur, Jr.
Deputy Director
Office of International Programs
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Lafleur:

REF: <u>Your letter 2/17/84, Review of NUREG-1061 Draft Report</u>

In accordance with your wishes, we have completed a review of NUREG-1061, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants," Pipe Crack Task Group, Draft Document. Our report is enclosed for your consideration.

Sincerely,

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AB Statens Anläggningsprovning
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APPENDIX H

REPORT OF INTERNATIONAL REVIEW TEAM

ON DRAFT OF NUREG-1061 - VOLUME I

INVESTIGATION AND EVALUATION OF STRESS-CORROSION CRACKING IN PIPING OF BOILING WATER REACTOR PLANTS

PIPE CRACK TASK GROUP

made for the U. S. Nuclear Regulatory Commission

by

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INTRODUCTION

The Pipe Crack Task Group have made an extensive appraisal of the intergranular stress corrosion cracking (IGSCC) problem in BWR welded Type 304 and 316 stainless steel pipework. This has resulted in a comprehensive report, NUREG-1061 (draft document completed March 1984), which we have reviewed in detail. In general, we find it to be a thorough investigation of the current status of the problem and the countermeasures which are in hand to alleviate the damage in operating plants and prevent its occurrence in new plants. The report is wide ranging in its coverage of the relevant issues, but we find that the conclusions and recommendations from individual sections have often not been sharpened up sufficiently. Hence, it does not quite provide as clear a quantitative basis for detailed guidance on future requirements as is possible on the basis of present evidence.

In our report below we have taken this into account and also drawn on our own experience of this phenomenon in Japan, Germany and Sweden. However, the views expressed in this report are personal and do not represent the official view of the regulatory body in our respective countries.

REPORT

Our report follows the order of conclusions and recommendations made in the Executive Summary of NUREG-1061, but we have tried to identify what we feel to be the six generic issues and made our comments in response to these. The issues are:

1) essence of the problem, 2) inspection, 3) flaw evaluation, 4) stress improvement techniques, 5) nuclear grade stainless steel, and 6) water chemistry options.

1. Essence of the Problem

We agree that the IGSCC problem is a result of a combination of factors: sensitized material, tensile stress (predominantly residual) and water environment. However, of the three factors, the material is the most important and a change of material to a nonsensitizing grade of austenitic steel (316NG, 347NG, 304NG) essentially removes the HAZ problem. We believe that if a new, adequate material replacement (in the old plants) or designation (for new plant) has been made, there is no need for a stringent requirement in regard to stress improvement measures or water chemistry treatment by hydrogen injection. The German experience with Type 347NG steel over many years and Japanese experience with Type 316NG steel over the past five years supports the view that material replacement effectively removes the problem. However, even when unsensitized material is used, it is prudent to adopt careful control of water chemistry by limiting conductivity to $<1 \mu S/cm$ and Cl^- to <0.1 ppm.

2. Inspection

Three generic problems must be recognized in the inspection of piping welds in existing plants for IGSCC:

- crack detection
- crack interpretation
- crack sizing.

All three are discussed at length in the Task Group Report.

We agree that crack detection and interpretation are considerably improved when procedures laid down in Code Case N-335 are followed and the operators are adequately trained on pipe welds containing IGSCC. It would undoubtedly be desirable to improve the validation of personnel, procedures and equipment even further and to extend the base of reference specimens with a larger variety of IGSCC. It must, however, be emphasized that numerous welds are accessible only from one side or not at all. For these, crack detection is extremely difficult or impossible. Therefore, full coverage with respect to detection cannot be obtained even under optimal conditions. A policy must be developed on a case-by-case basis for such welds in existing plants. This should take note of the feasibility of effective countermeasures such as replacement or stress improvement (e.g., IHSI). In new plants, the provision for a reduction in the number of welded joints and of access for weld inspection should be an obligatory design feature and not just an improvement option as stated in the recommendations of the Task Group.

We agree with the Task Group conclusions that crack length can be sized with sufficient accuracy for use in evaluation. Crack depth cannot, at present, be determined under field conditions with such degree of reliability that the results could be used as acceptance criteria. (NOTE - The comment on a 10% flaw depth acceptability in Section 5.2.3 is not justified in the light of depth sizing uncertainty.)

3. Flaw Evaluation

While agreeing with the Task Group's conclusions regarding the ability to size crack length with sufficient accuracy, we feel that more elaborate and stringent rules should be laid down on flaw acceptability. The 20% rule on tolerable crack length is evidently based on service experience with leaks and is supported by limited testing of flawed small diameter pipes and limited fracture mechanics evaluation. We agree with the Task Group Report that adherence to this rule should ensure a leak rather than a double-ended break. However, we are concerned about the potentially low fracture resistance of austenitic weld metal into which HAZ-initiated IGSCC cracks may be driven. For such cases on the large diameter pipes, we feel it is necessary to validate by full-scale pipe fracture testing the fracture mechanics assessment. Both static and dynamic testing should be considered.

The 20% rule should be taken as giving the maximum size of flaw which can normally be treated without immediate removal of the cracked section and replacement. In order to allow continued operation in the short term for flaws $<20\%$ circumference, countermeasures such as overlay welding or clamshells are acceptable. In the case of multiple cracks, it is recognized that they may readily link up. The ASME XI flaw characterization rules for multiple cracks should be used to define an "effective" crack length.

4. Stress Improvement Techniques

We recognize that the reduction of residual tensile stresses at the inner pipe wall to near or below zero is an effective countermeasure in preventing the initiation and growth of short cracks. This is achieved by IHSI and to a lesser extent by HSW. Residual stress can also be reduced to near zero by solution heat treatment (SHT), which also removes weld sensitization. Use of the latter is restricted, however, to shop-fabricated welds. The use of IHSI is supported as a preventative measure rather than a repair procedure. It should not be applied to pipes which are known to be cracked.

5. Nuclear Grade Stainless Steel

Type 347 modified nuclear grade material is used in Germany with a C content of <0.04%, S<0.02% and Nb<0.65%, but with a Nb minimum of 8X%C. The delta ferrite content (Delong) is 2-10%. With these requirements, the material is completely resistant to IGSCC and hot cracking on welding. No fabrication problems have been experienced in both shop and site fabrication. For 25 years, no IGSCC has been observed in this material. Type 316NG has been used in Japan for a number of years and more recently in the United States and Sweden. In Japan a level of N up to 0.12% is allowed with a limit on C+N of <0.13% and the CR and Ni equivalents are controlled to ensure a fully austenitic microstructure in the as-manufactured condition. After melting and resolidification, the material delta ferrite content is a few percent, which prevents hot cracking. Where a lower strength level is sufficient, Type 304NG with similar characteristics to those noted above has proved adequate in Japanese experience.

6. Water Chemistry

Possible modifications to the water chemistry, in order to inhibit IGSCC, are under consideration, in particular the injection of hydrogen to reduce oxygen level to 20 ppb. Hydrogen injection is still in an experimental stage and it remains to be verified that there are no detrimental side effects (e.g., on fuel). We agree with statements in the report concerning synergistic effects of oxygen and anion impurities. A limit should be put on conductivity (say ~0.2μS/cm) to warrant any conclusion that IGSCC will be prevented in sensitized material. We feel that at the present time the uncertainties related to hydrogen injection outweigh its benefits. However, it may be desirable to confirm it as an effective alternative countermeasure to stress improvement for cases where the replacement option is precluded. Start-up deaeration in order to reduce the oxygen content to values below 0.2 ppm is desirable.

CONCLUDING REMARKS

IGSCC in BWR piping systems deserves stringent attention to both the initiation and growth of cracks. Although it is anticipated that leak-before-break is inherent to tough, austenitic material, it is necessary to base the safety assessment for replacement of cracked sections and the criterion on a limitation of crack length. Current ultrasonic methods, properly applied, are capable of assuring the criterion of a limiting crack length. For welds which are not accessible to ultrasonic testing, from both sides, it is necessary to consider replacement or at least to apply adequate countermeasures against double-ended failure. Improved nuclear grade materials should be used for all replacements and any new construction.

ACKNOWLEDGEMENTS

The reviewers are grateful to Dr. S. H. Bush for his elucidation of the philosophy and content of the Task Group Report. They also appreciate the efficient secretarial service of Mrs. C. M. Bennett and the support provided by Mr. Howard Faulkner of NRC.

APPENDIX I

CONCLUSIONS AND RECOMMENDATIONS FROM NUREG-0531

The relevant conclusions and recommendations from the 1979 review, NUREG-0531, numbered as in the report, are given below.

2.10 Conclusions

- For operating plants and plants under review for an operating license or construction permit, the industry is implementing on a plant-specific basis many of the recommendations made by the previous Pipe Crack Study Group.

- The cracking experience in a German BWR facility demonstrates that IGSCC can occur in large-diameter (more than 20 in.) BWR stainless steel piping and confirms the previous study group's conclusion that, although cracking in large-diameter pipes has not been observed in the United States, the conditions for IGSCC do exist in these lines. We also have reviewed the safety aspects associated with the potential for IGSCC in large-diameter BWR stainless steel piping. Based on our review, we have concluded that, although throughwall and part-throughwall IGSCC can exist in BWR stainless steel piping during plant operation, it is unlikely that significant cracks would go undetected.

In addition, based on our review and analysis, we also conclude that it is unlikely these cracks will cause unstable crack growth and excessive loss of coolant. Further, since the ECCS provides protection should a loss-of-coolant accident occur, we conclude that IGSCC in these lines, while generally undesirable, will not be a hazard to public health and safety.

- There have been no IGSCC incidents in large-diameter (more than 20 in.) BWR stainless steel piping in Japan.

- Because significant IGSCC has occurred in recirculation-riser piping in Japan, we conclude that similar cracking may occur in plants in the United States.

2.11 Recommendations

- We recommend that the recommendations contained in NUREG-0313 continue to be implemented for operating plants and plants under review for an operating license or construction permit.

- Based on the incidence of IGSCC in recirculation-riser piping in Japan, we recommend that an augmented in-service inspection program should be developed for these lines. It is recommended that the augmented in-service inspection program conform to that recommended for nonconforming, service-sensitive lines in NUREG-0313.

4.7 Conclusions

- IGSCC can occur only if the proper combination of stress, environment and material conditions are present. Such a combination exists for austenitic stainless steels in BWRs, and in PWR systems other than the primary systems.

- Types 304 and 316 stainless steel in the sensitized conditions are susceptible to IGSCC. Safe ends of these materials, which were heavily sensitized by furnace heat treatment, have been cracked in BWRs. Piping of Type 304 stainless steel, lightly sensitized by welding, has been cracked in BWRs and PWRs. These materials in the annealed condition, or with low carbon levels, appear to be immune to IGSCC in oxygenated water, as do weld metal and cast metal of compositions similar to Type 304 stainless steel, but with a duplex structure containing ferrite.

- In both BWRs and PWRs, stress sources are not well defined, but residual stresses--and particularly residual stresses resulting from welding--appear to be major contributors to IGSCC.

- IGSCC is accelerated by cold working if the material is sensitized.

- IGSCC is accelerated by crevices or crevice-like conditions.

4.8 Recommendations

- Continue to make efforts to eliminate or minimize those conditions of stress, environment and materials that tend to cause IGSCC in BWR primary systems and PWR systems other than primary systems.

- The use of regular grades of Types 304 and 316 stainless steel in BWR piping systems should be avoided. If these materials are used, steps should be taken to ensure that IGSCC cannot occur. Such measures may include solution--annealing, weld cladding or other measures that have been demonstrated to eliminate sensitization and reduce residual stresses. Consideration should be given to techniques now being developed, such as IHSI, Heat Sink Welding (HSW), Solution Heat Treatment (SHT) of weldments, and Corrosion-Resistant Cladding (CRC), which are discussed in Chapter 10. In addition, tests should be run on each heat of piping to ensure that heats especially susceptible to sensitization are not used.

- Avoid cold working of austenitic stainless steels in LWR piping. All material should be in the annealed condition.

- Investigations of IGSCC to assess the effects of actual operating stress and thermal loading superimposed on constant loading in a manner that simulates reactor thermal cycles should be continued.

5.6 Conclusions

- Oxygen dissolved in the primary coolant is the principal environmental cause of IGSCC in BWR primary piping.

- Control of oxygen is highly desirable in BWRs. Oxygen control during shutdown and/or startup is considered feasible at the present time for most operating BWRs.

- The data do not exist at the present time to determine if oxygen control during shutdown and startup will prevent IGSCC in BWR piping.

- Deaeration of the condensate storage tank would have a beneficial effect by reducing the number of instances of cracking in the ECCS and CRC lines.

- The experimental techniques for controlling oxygen during steady-state operation, currently being developed under DOE sponsorship, may in the long run be more successful than simple control of oxygen during shutdown and startup alone.

- Contamination of the primary coolant in both BWRs and PWRs from resin bed intrusions and demineralizer breakthroughs in BWRs or from inadvertent opening of the boric acid or NaOH core spray lines in PWRs represents a definite potential cause of stress corrosion.

5.7 Recommendations

- Control of oxygen during shutdown and startup of BWRs should be practiced wherever possible.

- Reduction of oxygen levels in the coolant during steady-state operation of BWRs is desirable.

- Water chemistry excursions, either by accidental intrusion of resin beds in BWRs or of impurities from core spray coolants in PWRs, should be reduced to the extent practicable.

6.6 Conclusions

- Piping systems in BWRs are complex structures containing many welds. From a stress analysis standpoint, each weld is itself a complex structure. Piping design codes provide upper limits to calculated operation stresses but, because those stress bounds exceed the material yield strength, the code stress limits are not appropriate for controlling the operation stress contributions to IGSCC. Fabrication (residual) stresses in stainless steel pipe are not controlled by piping code rules.

- Welding residual stresses (not followed by annealing) are high (e.g., 25 to 60 ksi). These are axial tensile stresses on the inside surface and possibly are a major contributor to IGSCC. The available test data suggest that residual stresses are higher in small Schedule 80 pipe than in large Schedule 80 pipe; this may explain in part why there is relatively more cracking in small pipes than in large pipes.

- The General Electric Design Stress Rule is a potentially useful tool, particularly in focusing in-service inspection on welds which, from a stress standpoint, would be more susceptible to IGSCC.

6.7 Recommendations

- It is recommended that work be continued to quantify residual stresses that are due to welding, both at pipe-to-pipe welds and pipe-to-fitting or valve welds. It is also recommended that investigation of procedures to reduce tensile residual stresses on the inside surface, such as those discussed in Chapter 6, be continued.

- It is recommended that a review of stresses in BWR piping systems be undertaken to establish the operating stress levels as a function of pipe size. The purpose would be to see if operating stress averages help to explain why there is relatively more cracking in small pipes than in large pipes.

- It is recommended that work continue to improve and apply the Design Stress Rule.

- It is recommended that grinding of the interior surface of weld regions be avoided under conditions where IGSCC may be present.

8.5 Conclusions

- The ASME B&PV Code's Sections V and XI ultrasonic examination procedures for ferritic weldments are not adequate for the detection and evaluation of IGSCC in austenitic piping.

- Signifcant IGSCC will be detected and evaluated reliably, provided the improved conventional UT techniques and procedures of recent years are followed.

- Quantitative sizing of IGSCC by amplitude measurements is considered unreliable.

- Difficulty of detection and evaluation of IGSCC by UT leads to ambiguous results; as a result, a small number of IGSCC will be either missed or called out incorrectly.

- The current leak detection systems installed in BWRs provide a reliable complementary means by which to detect cracks. Experience indicates that these leak detection systems will detect a crack before it propagates to a size that would result in a significant loss of coolant.

- Plant shutdown should be initiated for inspection and corrective action when the leakage system indicates, within a period of four hours or less, an increase in the rate of unidentified leakage in excess of 2 gpm, or when the total unidentified leakage attains a rate of 5 gpm, whichever occurs first.

8.6 Recommendations

The Pipe Crack Study Group recognizes that the following recommendations are being pursued to a substantial degree and that definite progress is being made:

- For conventional UT testing methods, all reflectors should be mapped with respect to weld geometry. Reflectors within the HAZ should be classified as IGSCC.

- As a minimum, implement a recording procedure by which amplitude and metal path are recorded automatically. Positional data of the search unit may be recorded manually on the amplitude and metal path data record, or if possible, automatically. Analysis of the UT data can then be done by independent examiners under favorable working conditions.

- Advanced detection and evaluation methods are being developed under several industry and NRC programs. It is recommended that those methods particularly useful for examining austenitic materials for IGSCC be pursued as actively as possible. The following areas should be given special emphasis:

  - Development and application of methods using advanced signal processing techniques to reduce dependence on amplitude and aid in differentiating between IGSCC and other reflectors. These include synthetic aperture focusing and adaptive learning network approaches.

  - Development and optimization of ultrasonic probes and acoustic parameters to enhance through-weld inspections and interpretation of signals, such as focused pitch-catch, phased arrays, variable frequencies, and variable angles.

- Development and testing of acoustic emission techniques specifically for detecting IGSCC in austenitic materials, with special emphasis on weldments. Any additional work required to apply acoustic emission to leak detection should also be pursued.

- Prepare specific procedures in the ASME Code to incorporate the improvements in detection and evaluation methods that have been developed to date. Issue these in the form of a Regulatory Guide pending either a Code change or an enabling Code Case to make them applicable.

- Focus inspection efforts upon those pipe welds most likely to crack. This could reduce the number of welds to be inspected, permitting more time to be spent on welds most likely to crack without increasing total personnel radiation exposure or plant down-time. The use of General Electric's "Stress Rule" Index to select welds for inspection should be evaluated as to its applicability in defining the welds most likely to crack.

- Complete work on ISA Standard S 67.03, ANSI Standard N 41.21 entitled, Standard for Light Water Power Reactor Coolant Pressure Boundary Leak Detection, and promote serious consideration by the NRC of its adoption.

9.3 Conclusion

From our review of IGSCC incidents in BWRs and our evaluation of the potential for unstable crack growth, we conclude that it is unlikely that these cracks will cause unstable crack growth and excessive loss of coolant. Further, since the ECCS provides protection should a loss-of-coolant accident occur, we conclude that IGSCC, while generally undesirable, is not a hazard to public health and safety. The following considerations are the bases for our conclusion:

- The IGSCC occurs in a small percentage of the welds in BWR piping and is likely to be detected before it grows to significant size.

- Even considering the most severe known IGSCC, large margins against unstable crack growth are maintained during normal operation and anticipated transients. Unanticipated transients that may be postulated to cause unstable crack growth prior to detection of the most severe flaws are low-probability events.

- A lost-of-coolant accident is a design basis for nuclear facilities in the United States.

10.6 Conclusions

From the limited information discussed above and in Appendix C, the following tentative conclusions can be drawn:

- Significant variability exists in different heats of commercial Type 304 stainless steel with respect to sensitization in the as-received condition. Also, a significant variability exists in the response of different piping heats to weld sensitization.

- There may be no significant difference in the sensitization behavior of 4-, 10-, and 26-in.-diameter Schedule 80 stainless steel piping when welded according to typical welding practices. All pipes become sensitized.

- The degree of sensitization in the HAZ of certain weldments could increase significantly after 10-year exposure at reactor operating temperatures.

- The grain boundaries of sensitized Type 304 stainless steel piping contain less chromium than is required for corrosion resistance. Also, sulfur and phosphorus concentrate at the grain boundaries where they can affect sensitization, mechanical properties, and the electrochemical characteristics of the grain boundary areas.

- The electrochemical potentiokinetic reactivation technique can distinguish between Type 304 stainless steel piping materials that are susceptible and nonsusceptible to IGSCC, both in the as-received or as-welded conditions. The technique can thus be used to verify acceptability of as-received material, of weldments, and of thermally treated components.

- Bulk residual stresses on the inside surfaces of welded Type 304 stainless steel Schedule 80 piping appear to be lower in 26-in. pipes than in 4- and 10-in.-diameter pipes. However, these stresses are considerably above the 550°F (287°C) yield strength of the base material for all pipe sizes.

- Welded Type 304 stainless steel pipes are more prone to sensitization than would be expected based on total time at sensitizing temperatures alone because of the synergistic effect between the thermal and strain cycles.

- The effects of continuous cooling on sensitization, as from welding, are more severe than the effects of isothermal treatments.

- IHSI may prove effective in redistributing the welding residual stresses in BWR piping, so that the pipe's inside surface is left in a favorable state of axial compression.

- For repair welds and welds on new pipes made of Type 304 stainless steel, several techniques have been identified to reduce the potential for IGSCC. SHT eliminates sensitization but is restricted to shop welds. A CRC with duplex microstructure can be weld-deposited to minimize the HAZ on the pipe's inside surface to move the HAZ away from the high-stressed region next to the butt weld, and to isolate the butt weld from the environment. A water spray on the pipe's inner surface during welding can produce a HSW with a compressive residual stress on the pipe's inner surface. Finally, tighter welding specifications to limit the amount of post-weld grinding will reduce the source of high residual stresses.

10.7 Recommendations

- It is recommended that studies be continued to understand the heat-to-heat variability of Type 304 stainless steel piping with respect to its susceptibility to IGSCC. Since a large variability exists, it is recommended that sensitization tests correlated to IGSCC susceptibility in BWRs be conducted for acceptance of piping material. Further, it is recommended that such tests be used for qualification of welding procedures and thermal treatments for each heat of piping material. The tests should be performed on each heat and/or product form.

- It is recommended that, when Type 304 stainless steel piping is removed from long-time service, it should be investigated to determine the presence of low temperature sensitization (LTS).

- It is recommended that studies be conducted to determine the effectiveness of IHSI on pipes with pre-existing flaws and on large-diameter (thick-walled) pipes, and to determine if IHSI induces or accelerates LTS.

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12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

Severe intergranular stress corrosion cracking (IGSCC) of the recirculation piping system in several boiling water reactors occurred during 1982-1983. A Task Group on Pipe Cracks was established by the U. S. Nuclear Regulatory Commission with the broad charter of developing in integrated program to deal with the entirety of the stress corrosion cracking problem. This report presents specific conclusions and recommendations that are tied closely to relevant regulatory documents so that necessary changes can be implemented.

This report covers aspects such as the causes and descriptions of IGSCC phenomena; current status of pipe cracking in BWR's; nondestructive evaluations of piping welds; inspection of piping for IGSCC; decisions and criteria for replacement, repaid of continued operation without repair; risks related to the presence of IGSCC; and a value-impact assessment of IGSCC.

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